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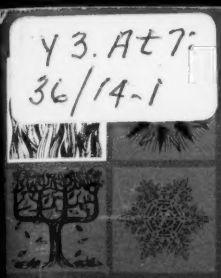

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# Reactor Technology

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# REACTOR TECHNOLOGY

Vol. 14, No. 1 Spring 1971

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# REACTOR TECHNOLOGY

is a quarterly review of progress and developments in the reactor field including the following subject areas:

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| Economics | Fuel Cycles |
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The many reports and publications referenced merit further study, and readers are urged to consult them for additional details.

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# Reactor-Materials Inspection by Neutron Radiography

By A. M. Plummer*

## INTRODUCTION

Neutron radiography is a relatively new method of nondestructive testing; already it has been found useful for a number of applications, and new ones are being tested daily. With most new technologies, investigators seeking applications are faced with the often difficult problem of matching the capabilities of their technology to others' needs, and neutron radiographers find their field no exception. This article, however, deals with one of the easy applications—reactor-fuel technology.

In most respects, neutron radiography is naturally and ideally suited to the examination of reactor fuels: First, of course, is the fact that, in general, fuels are neutron absorbers and the materials that surround them are not. Second, the method first used (and still popular) for neutron imaging is selective for neutrons and is insensitive to ionizing radiations. With these characteristics, neutron radiography can be applied in a straightforward manner to the nondestructive examination of many types of irradiated, highly radioactive fuels. The examples cited in this article have been selected to show, among other things, how neutron radiography has been used to determine the progress of changes that occur over the irradiation period of experimental fuels. An obvious and extremely important advantage of such progressive surveillance is that it provides information that assists the experimenter in judging the best time to terminate an experiment for specialized destructive or nondestructive postirradiation examinations.

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## COMPONENT REQUIREMENTS AND METHODS

Three basic components are required to perform neutron radiography: a neutron source, a neutron collimating system, and a neutron-beam imaging device. Several alternatives exist in the choice of each of these components, but the functions of the components are not entirely independent, and their selection must be based on several factors, among which are the nature of specimen itself and the information that is sought.

### Sources

Neutron radiography can be performed with reactor, isotopic, or accelerator sources, but, at the present time, a reactor is the best neutron source to use for fuel examinations simply because it provides the highest intensity of the right kind of neutrons. Interaction cross sections are generally highest for thermal or near-thermal neutrons, and consequently this energy range provides the greatest contrast for most materials combinations. However, for some special purposes (including reactor-fuels examination), there may be an advantage in modifying the neutron spectrum to take advantage of higher cross sections in the resonance or subthermal energy ranges. In any case, the neutrons that can best be used for radiography are present in any thermal reactor.

### Collimators

The term "collimator" is somewhat of a misnomer as it is used in neutron radiography because in most cases the collimator does not actually cause all neu-

trons in the beam to move in parallel paths. The principal function of this component is to limit the size of the neutron source and to allow neutrons from the thus-defined source region to pass to the specimen unattenuated by the biological shielding that ordinarily surrounds the source. Thermal-neutron sources, incidentally, are necessarily distributed sources because neutrons of this energy can be produced only through scattering collisions; thus radiography with thermal neutrons requires the use of a collimator.

In actuality the collimator is a shielded aperture that defines the source of a more or less diverging neutron beam. The figure of merit for collimators is the ratio of source-to-specimen distance to aperture diameter ($L/D$). This ratio is a measure of the angle subtended by the source at the specimen position and is a parameter of particular significance in the evaluation of neutron-radiography facilities. Note that the size of the exit aperture of the collimator defines only the effective size of the available neutron beam and has no bearing on the critical geometrical parameters of the collimator.

The $L/D$ ratio determines in part the sharpness of the images obtained in a neutron radiograph. The importance of this effect is seen in Figs. 1 and 2. Figure 1 shows schematically how the image of a cylindrical object (relative size greatly exaggerated) is broadened by the physical extent of a distributed source; note that the specimen is located at some

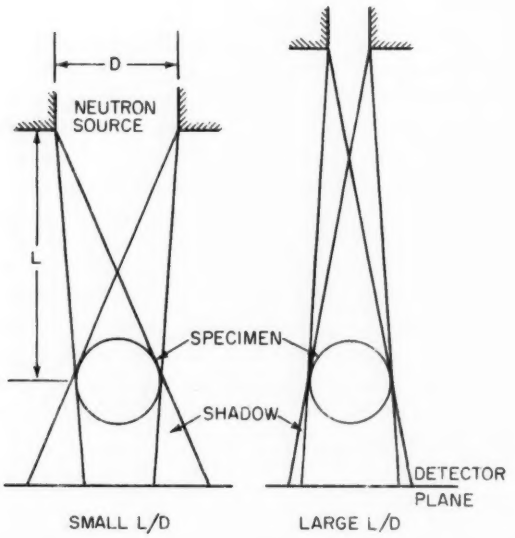


**Fig. 1 Schematic representation of edge image obtained with finite-sized source.**

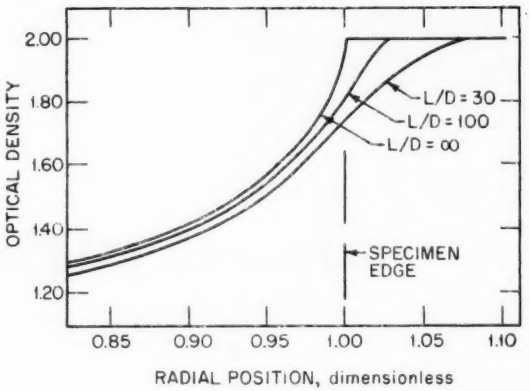


**Fig. 2 Effect of $L/D$ ratio on image sharpness.**

distance from the detector plane, as it would be if the specimen were located inside an irradiation capsule. Figure 2 shows plots of optical densities that would be obtained in a typical radiograph of a cylindrical fuel specimen under several conditions. The $L/D$ ratios of 30 and 100 are representative of isotope and reactor facilities, respectively; the infinite ratio is that of a truly collimated source. The optical-density values plotted in Fig. 2 include exposure due to neutron transmission through the specimen, as well as the effects of the source size and distance from the cylindrical specimen. Note that an optical density of 2.00 is produced by an unattenuated neutron beam striking the detector. These effects are important not only in the case of edge definition but also in the sharpness of the image of other features of the specimen as well. Clearly a large $L/D$ ratio is necessary if neutron radiography is to be used to detect cracking, void formation, and similar small defects in experimental fuel specimens.

## Imaging Methods

Like X-rays, neutrons do not produce direct ionization in their passage through materials; therefore products of their interactions must be utilized to produce images. In practice, two types of interactions are used: those which yield prompt charged particles and those which produce charged particles through the radioactive decay of reaction products. Imaging methods associated with these reactions are known, respectively, as direct and indirect methods.

In using the direct method, an image-recording medium and a neutron-to-charged particle "conversion screen" are placed together in the neutron beam to

intercept and record the information imparted to the beam by the specimen being examined. Among the advantages of the direct method is its ability to provide both integrated and real-time images, depending on the choice of converter and detector. For example, information can be accumulated over relatively long periods of time by using a photographic emulsion to record either charged particles from a gadolinium-foil converter or photons from a neutron scintillator. By various means the radiations from the same conversion reactions can provide a real-time video presentation of the information-carrying intensity pattern. The direct method has the disadvantage that in any of its variations it is sensitive to gamma radiation; thus the usefulness of this method is extremely limited in its application to irradiated-fuel examination and, for all practical purposes, is restricted to preirradiation examinations.

The neutron-radiography method best applied to radioactive materials in general, and to irradiated fuels in particular, is the so-called indirect method. In this method the information-carrying neutron beam is allowed to activate the conversion medium, and the image so obtained is transferred by the charged particles of radioactive decay to the image-recording medium (usually a photographic or X-ray emulsion). The resulting final image is thus an autoradiograph of the neutron-activated conversion screen. This method is also known appropriately as the transfer technique.

As mentioned earlier, the indirect method is ideally suited to the examination of highly radioactive fuel materials because of its inherent discrimination against gamma radiation. By the use of this type of neutron radiography, irradiated specimens can be subjected to a detailed nondestructive examination within a matter of a few hours after their removal from irradiation facilities. The full significance of such a capability is probably best appreciated by the experimenter who has terminated an experiment on the basis of irradiation time—only to find from subsequent destructive examination that he has done so either too early or too late.

## APPLICATIONS IN FUEL-CAPSULE SURVEILLANCE

Neutron radiography is used routinely at several facilities as an inspection tool in fuel-development and -evaluation programs. At the Battelle Columbus Laboratories, for example, fuel-containing irradiation capsules installed in the Battelle Research Reactor (BRR) are examined by neutron radiography prior to irradiation and at the end of each operating cycle, i.e., at 3-week intervals. Swimming-pool reactors are particularly suited to the use of neutron radiography for this application; the open pool permits easy access to experiments and a minimum of handling of specimens. The radiography facilities can be located in the pool adjacent to the reactor, so that it is possible to perform all capsule manipulations underwater with no requirement for cask transfer or facility shielding. The relative ease with which these operations can be performed makes it possible, both operationally and economically, to use neutron radiography for a variety of purposes.

At the BRR, fuel-capsule surveillance begins with a preirradiation neutron radiograph. This radiograph provides the reference information for subsequent examinations and also supplements preirradiation X-radiographs, which are made to verify proper capsule assembly (the neutron radiograph can verify the composition of the fuel as well as its location). Subsequent neutron radiographs are then normally made at 3-week intervals, as mentioned, to provide the information that is required for the particular experiment. Normal surveillance examinations are planned to follow the progress of expected or observed changes in the physical composition of the fuel specimens. Although these examinations are intended to provide experimental data, they can also yield information to assist in the diagnosis of malfunctions and, from a safety standpoint, provide a record of capsule integrity.

Experiment-related information from surveillance radiographs varies widely and is best illustrated by particular examples. These examples have been selected to show easily seen gross degradation of the fuel specimens, but the neutron radiographs are also used to determine and measure much more subtle changes. The specimens shown in Figs. 3 to 8 are ceramic fuels from a high-temperature fuels-evaluation program that involved the irradiation of a total of 87 specimens in 15 separate capsule experiments. Neutron radiographs of two orientations were made at regular intervals throughout the irradiation periods, and the images were measured to determine dimensional changes in the specimens. Volume changes calculated from these measurements were used in many cases to predict subsequent degradation long before the damage itself could be observed.

Figures 3 to 8 show sets of radiographs of individual specimens from two different capsule experiments (note that the figures are reversals of the radiographs). Each set includes the preirradiation radiograph and the radiograph after the third reactor

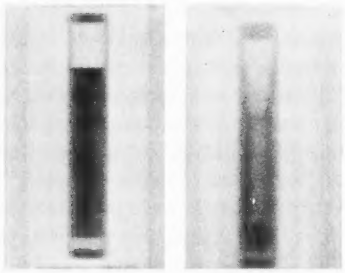


Fig. 3 Neutron radiograph of fuel specimen irradiated at a surface temperature of 1700°C. (Note centerline void formation and fuel migration.)

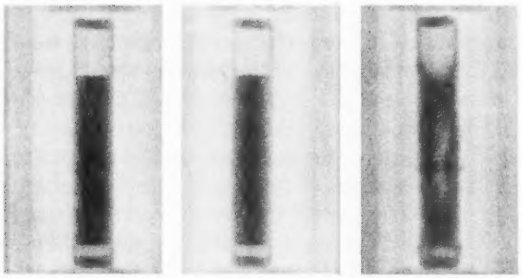


Fig. 4 Neutron radiograph of fuel specimen irradiated at a surface temperature of 1750°C. (Note enclosed centerline void.)

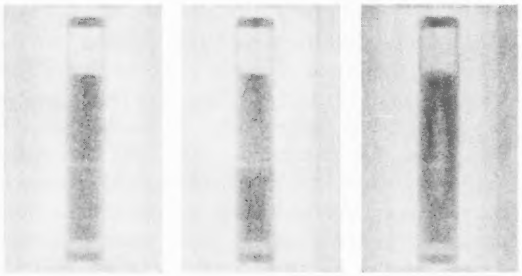


Fig. 5 Neutron radiograph of fuel specimen irradiated at a surface temperature of 1700°C. (Note void formation without elongation.)

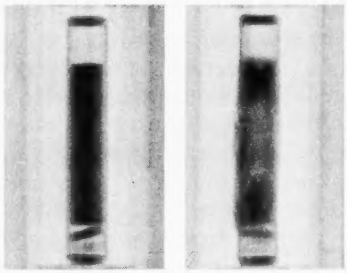


Fig. 6 Neutron radiograph of fuel specimen irradiated at a surface temperature of 1750°C. (Note void formation and radial swelling.)

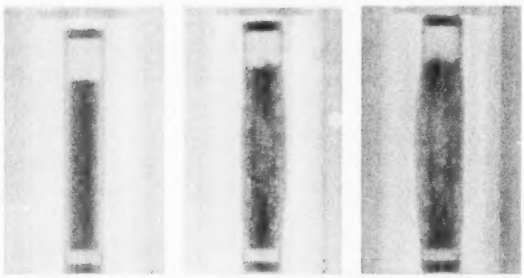


Fig. 7 Neutron radiograph of fuel specimen irradiated at a surface temperature of 1750°C. (Note elongation and radial swelling.)

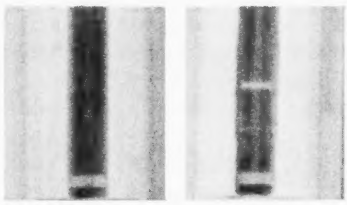


Fig. 8 Neutron radiograph of fuel specimen irradiated at a surface temperature of 1650°C. (Note pellet segregation.)

operating cycle; Figs. 4, 5, and 7 also include the radiograph following the sixth cycle. Each operating cycle lasts about 430 hr, providing about $1.5 \times 10^{19}$ neutrons/cm$^2$ in the core position occupied by the capsules.

Figure 3*b* shows a large centerline void; the apparent migration of the fuel up the inside of the cladding wall indicates that vaporization and redeposition of the fuel occurred in this case. Figure 4*c* shows similar behavior, of a different specimen, but here the void formed after a longer irradiation exposure and is somewhat different in nature. The void in Fig. 4*c* appears to be totally enclosed, although the presence of fuel on the upper cladding wall indicates that some vaporization has occurred.

Figure 5*c* also shows the formation of a centerline void, but, unlike the previous examples, there is very little indication of vaporization and upward redistribution of the fuel. The appearance in this case suggests that there was migration of the fuel within the matrix to the outer regions of the specimen as the fuel densified and that there was relatively little change in the specimen volume.

Figures 6*b* and 7*b* and *c* show small, more uniformly distributed voids accompanied by specimen radial swelling. Figures 7*b* and *c* show evidence that elongation of the specimen containment occurred along with the radial swelling. This latter information is an example of the kind that dictates termination of the experiment to preserve the specimens for subsequent examination; in this case continued operation of the capsule would quite likely cause physical damage to this specimen (and possibly others) because of interference with the baffle plates between adjacent specimens inside the capsule.

Figure 8 shows the separation of a pelleted specimen caused by axial shrinkage.

Information of the type shown in Figs. 3 to 8 can be extremely valuable in warning experimenters if something is amiss in time for them to stop the test and preserve the specimens. In addition, such periodic inspections can provide insights into the time-dependency of degradation that can be used to postulate the mechanism and consequences of irradiation and temperature-related damage.

A different type of examination is illustrated in the neutron radiograph of Fig. 9. Here the purpose was to try to determine the cause of an electric-heater failure that occurred during capsule operation. The bead formed by melting of the wires as the result of a short circuit at the heater-lead-wire junction can be seen (arrow) in the annular space on the right side of the capsule interior. This information was the deciding factor in the decision that repair of the capsule was impractical.

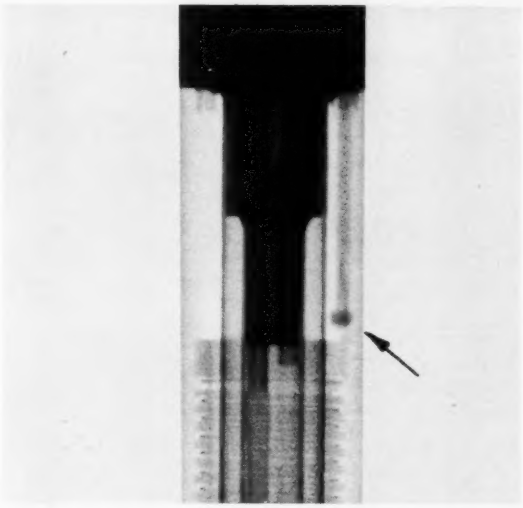

**Fig. 9 Neutron radiograph used to locate an electric-heater failure (arrow) in a fuel-irradiation capsule.**

Figure 10 shows a portion of a neutron radiograph of several experimental fast reactor fuel rods. The rods were subjected to similar irradiation exposures but are encapsulated in different physical environments; pellet cracking is evident in some cases. All these rods are being subjected to extensive postirradiation examinations which will yield information that is vital to the development program. However, if this were simply a screening test, it is apparent that the use of neutron radiography could eliminate much of the nondestructive examination that would be required to select acceptable configurations.

## SUMMARY

The selected examples of neutron radiography presented show how this new technology is used in the routine surveillance of reactor-fuel-irradiation experiments. There are other applications—some related to reactor fuel and some not; some routine, and some experimental—and many more will surely be found. But, whatever its future adaptations, neutron radiography has already become firmly established as a unique source of information in the evaluation of reactor-fuel performance.

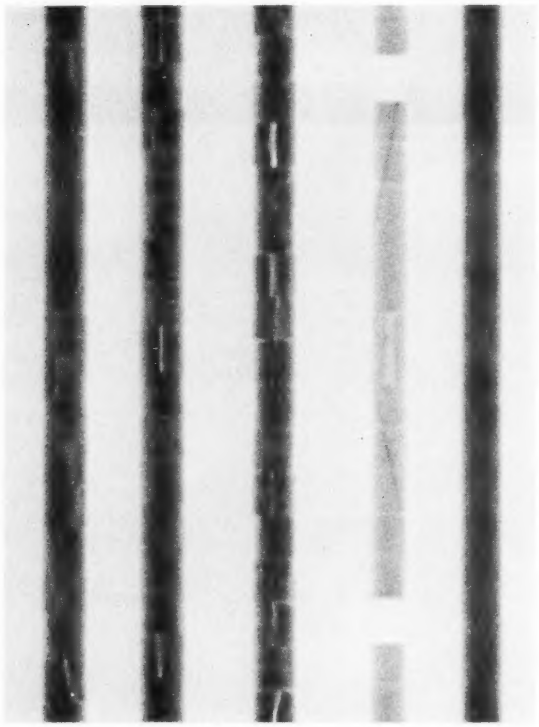

Fig. 10 Neutron radiograph of five experimental fast reactor fuel rods. (Note pellet cracking.)

# Operating Experience at the Connecticut Yankee Reactor

By Myrna L. Steele*

The Connecticut Yankee Reactor, located on the east bank of the Connecticut River about 1 mile from Haddam and 21 miles south-southeast of Hartford, is a 600-MW(e) pressurized-water reactor (PWR) owned by the Connecticut Yankee Atomic Power Company. The reactor achieved criticality[1] on July 24, 1967, and reached initial design power in December 1967. On Jan. 1, 1968, commercial operation was begun. On Mar. 11, 1969, an increase in gross power level to 600 MW(e) was authorized, and 1 week later the reactor output was a net 575 MW(e). A scheduled 1-month shutdown for maintenance began Apr. 11, 1969. When the reactor was restarted, it was operated at various power levels up to 600 MW(e) until the refueling shutdown that lasted from Apr. 17 to June 24, 1970. By Sept. 30, 1970, the reactor had produced about $9.5 \times 10^6$ MWh (net) of electricity with an availability of about 80% during more than 3 years of operation.

The cylindrically shaped core lattice[2] is fueled with slightly enriched $UO_2$ pellets clad in stainless-steel tubes and has an active height of 10.15 ft and a diameter of about 9.97 ft. Control of the reactor is achieved by cylindrical absorber rods (called rod-cluster control assemblies) that are inserted directly into guide thimbles within some of the fuel assemblies. The 157 fuel assemblies comprising the core are positioned by upper and lower grid plates and are surrounded by a form-fitting coolant-flow baffle and then by a core barrel. Long-term reactivity effects in the four-loop closed cooling system are controlled by varying the boric acid content in the cooling water.

*U. S. Atomic Energy Commission, Division of Technical Information Extension, Oak Ridge, Tenn. 37830.

The turbine cycle[3] consists of the turbine generator, deaerating condenser, condensate and feedwater systems, and turbine-generator auxiliaries. The turbine rating is 612 MW at an absolute exhaust pressure of 2 in. Hg and at a throttle steam condition of $7.67 \times 10^6$ lb steam/hr, with steam being supplied ahead of the main stop valves at 630 psia, 490.5°F, and 0.25% moisture. The generator is rated at 667,000 kVA with a hydrogen pressure of 60 psig.

## PRESTARTUP OPERATIONS

By July 1, 1967, final preparations[4] were under way for core loading. Functional checkouts of the fuel-handling system, rod-cluster change fixture, pressure vessel, and limit-switch and interlock settings were completed by July 2. At 2100 hr on July 2, the reactor vessel was filled with borated water; one of the neutron sources was loaded into the vessel, and core-loading activities were begun. These activities were interrupted periodically to allow final inspection of the reactor internals and were continued until 0600 hr on July 6. On July 15 the cooling system was filled, vented, and brought to 250°F for cold-condition control-rod-drop tests. The cold tests were completed on July 19, and rod-drop tests for the hot reactor condition were finished on July 20. On July 21, leakage at two feedwater check-valve flanges, on two nonreturn valve bypass lines, and on one pressurizer safety valve caused the reactor systems to be cooled down for repairs. These repairs were completed by July 23, the system was brought to 250°F, and the approach to initial criticality was begun at 2100 hr by boron dilution. Criticality was achieved at 0104 hr on July 24. Acquisition of cold physics data and hot low-power testing

data was completed by July 30. On July 31 the initial attempt to roll the turbine failed because excessive control-valve leakage limited turbine control and speed.

During the low-power physics testing, two unscheduled scrams occurred. On July 25 an inadvertent actuation of an overvoltage relay in the rod power supply caused an automatic reactor trip from >1 MW(t), and on July 28 an automatic scram from >1 MW(t) was initiated by a noise spike in an intermediate-range control circuit.

## LOW-POWER AND APPROACH-TO-POWER OPERATIONS

Efforts[5] to provide for pressure equalization so that the turbine could be rolled resulted in the discovery of an oil leak within the oil reservoir (which was repaired) and the subsequent deduction that the output of the auxiliary oil pump was inadequate. As a consequence the steam-generator pressure was reduced to 400 psig, pressure equalization was achieved, and a turbine roll was accomplished at 1800 rpm on August 4. At 1905 hr on August 7, the generator was phased to the 345-kV grid, and the plant was loaded to 50 MW(e). At 1935 hr the power run was terminated because of erratic feedwater control, excessive leakage on the bypasses for the nonreturn valves, loss of level control on the heater drain tank, and partial plugging of the suction strainer for the condensate pump. Plant cooldown was begun immediately after the reactor scram so that the necessary repairs could be made to the bypasses for the nonreturn valves. During cooldown, an inspection of the primary-coolant-pump motors revealed a partial failure of some of the antireverse rotation pawls. New pawls were procured and installed.

After additional inspections, plant heatup was initiated on August 13, and the 525°F level was reached early on August 14. During heatup the electrohydraulic operator on one of the two pressurizer spray valves failed. Continued difficulties with the hydraulic operators throughout August prompted the decision to replace the electrohydraulic valve operators with air operators. Two small air compressors were installed in the containment building specifically for this purpose since plant instrument air is not used in the containment except during shutdown.

Criticality was again achieved at 2025 hr on August 14, and the plant was synchronized to the line and loaded to 50 MW(e) at 0134 hr on August 15. At 0305 hr a turbine trip was initiated from the transformer-supply breaker on a station-service transformer that feeds the primary-system coolant pumps. An attempt to start one of the pumps had resulted in a noise spike that exceeded the overcurrent set point on the transformer. At 0531 hr the plant was returned to the line and was operated at 50 MW(e) for 8 hr prior to overspeed trip testing of the turbine. The plant was removed from the line at 1336 hr, and, as the power was being decreased, erratic behavior of one of the pressurizer spray valves and one of the motor-operated feedwater isolation valves necessitated a reactor scram from 10 MW(e). The pressurizer spray valve had opened, for no apparent reason, at 25 MW(e) and could not be closed until primary-system pressure had decreased to 1850 psig. The difficulty with the motor-operated feedwater isolation valve stemmed from improper setting of the torque limit switches that were inspected and reset subsequent to reactor shutdown.

On August 16 at 2241 hr, the turbine-overspeed testing was completed, and the plant was synchronized to the line and loaded to 30 MW(e). At 2301 hr, as the power level was being increased to 50 MW(e), automatic scram at about 35 MW(e) was actuated by a full-open limit-switch signal from the main steam isolation valve. A solenoid-valve failure had allowed the set point of the first-stage pressure indication for the turbine to slip downward, and thus a full-open limit-switch signal was generated. The plant was returned to the line at 0040 hr on August 17 and was tripped off-line again at 1348 hr by an unintentional grounding of the vital bus during calibration of the reactor control and protection system. The plant was returned to the line later in the day.

When lightning struck the 345-kV line on August 18, it caused an automatic turbine trip. Subsequent line inspection revealed no damage, and the turbine was rolled again. As the turbine speed reached 1800 rpm, a high-vibration alarm was annunciated from a bearing at the exciter end of the turbine. Turbine speed was immediately reduced, manufacturer's representatives were called in, and a detailed inspection was carried out. No abnormal conditions were evident. Several balance runs were made, and the unit was synchronized to the line on August 19 at 2304 hr. At 20 MW(e) a slight increase in vibration was noted; at 30 MW(e) the vibration exceeded the 5-mil alarm set point; at 40 MW(e) the vibration had increased to 8 mils, and the unit was shut down for more balance runs. The turbine was returned to the line on August 20 at 0706 hr.

Leakage of the pressurizer safety valves was noted first on August 18. By August 21 the feed-and-bleed

rate of the pressurizer relief tank was approaching the capacity of the boron-recovery system therefore the plant was taken off-line, and plant cooldown was initiated at 2208 hr on August 21. By August 23 the primary system had been cooled down to 130°F, and repairs to the pressurizer safety valves were begun. The hydraulic operators on the pressurizer spray valves were replaced with air operators, and a failed seal on one of the coolant pumps was replaced. The vessel head and the upper part of the core internals were removed on August 31 in preparation for the replacement of all in-core thermocouples. Thermocouple repairs[6] had been completed by September 10, and the upper internals and vessel head had been replaced.

By September 16 the primary system was filled and vented, and heatup was initiated on September 19. The plant was placed in three-loop operation and the reactor brought critical on September 20 for reactor-operator examinations. As the power was being reduced after the operator examinations, instrumentation personnel, who were calibrating the first-stage pressure circuit of the turbine, unintentionally cleared the 10% of rated power permissive circuit for the turbine and scrammed the reactor. On September 22 the reactor was again brought critical, and secondary-system heatup was begun. An automatic scram from <1 MW(t) was initiated by a steam-flow–feedwater-flow mismatch caused by opening of the steam-dump valves after clearance of the low-vacuum interlock. Subsequent inspection of the steam-dump control circuit showed that a locked-in test signal, which had been installed during shutdown for control-system calibration, had not been removed. Once the low-vacuum block was cleared, the steam dump occurred automatically when the secondary-system temperature approached the dump set point. Late that day primary-system cooldown was initiated to repair leaking safety valves and solenoid-type relief valves. Also, during the shutdown the packing on both pressurizer spray valves was repaired, the coupling on one of the reactor coolant pumps was realigned, and two of the main steam safety valves were repaired.

The primary system again was filled on September 25, and heatup was begun. By the end of September, the plant was in the hot-standby condition.

On October 2 the secondary system[7] was heated, the turbine was started, and the generator was phased. Shortly thereafter a gasket blew on one of the turbine-stop-valve pressure-equalizing lines; the generator was unloaded from the line, the turbine was shut down, and the steam was removed from the secondary system. When the gasket was repaired, the secondary plant was restarted, and the generator was phased to the line. After an 8-hr calorimetric test was completed at 50 MW(e), the power level was increased to 82 MW(e), and a similar test was conducted at that level. On October 4 the power level was increased to 125 MW(e). An automatic reactor and turbine trip from 125 MW(e) occurred at 0701 hr on October 5 when a vital-bus power supply was interrupted. After the power to all primary-system and nuclear instrumentation was restored, the steam was removed from the secondary lines in preparation for repairs in the secondary system. By October 10 the repairs were completed, steam was returned to the secondary system, and the turbine was rolled. After some minor problems with the turbine were cleared, the generator was phased to the line, and the power output was increased to 80 MW(e). Subsequently three bus-transfer tests were completed successfully, and the power level was increased to 122 MW(e) for about 3 hr. A steam leak at a control valve in one of the lead-in steam lines to the high-pressure turbine necessitated a power reduction for repairs. During the power reduction, a relay that had received a rapid-load-rejection signal tripped and caused a turbine trip–reactor scram. By October 12 the lead-in steam line was repaired, and the generator was phased to the line and loaded to 122 MW(e).

Problems with level control in the reheater-drain tank required reduction of generator load to 39 MW(e) for repairs. While the reactor was at the 39-MW(e) level, the No. 1 inverter blew a fuse and caused a reactor scram. After the power had been restored to the primary-system and nuclear instrumentation, repair work was begun on the secondary system. One condenser tube was plugged, the reheater control valves were disassembled, and 10 to 20 mils were removed from the body bushings, and the manual isolation valve was overhauled. When the repairs were completed, the reactor was started up and shortly thereafter was tripped by a spurious signal. Another startup was immediately initiated, the secondary system was warmed up, and the generator was phased and loaded to 170 MW(e). The next day an automatic reactor trip occurred because of an unintentional grounding of the vital bus. An immediate restart was effected, and the generator was phased and loaded to 250 MW(e) where it stayed for about 9 days. On October 25 the power supply to the coincidenter blew a fuse, indicated a high intermediate-range startup-rate signal, and scrammed the reactor. Steam was removed from the secondary system until October 29. During the plant shutdown, some in-core zero-power flux traces were made, and,

since the coolant temperature was maintained at 550°F, moderator temperature-coefficient data were collected. On October 30 the generator was phased to the line and loaded to 250 MW(e). Difficulties with control valves necessitated removal of the plant from the line. All four valves were disassembled for inspection, which revealed missing stem-locking pins at the lower bushings in two of the control valves. The plant remained in the hot-shutdown condition through November 2.

On November 2 the secondary plant[8] was warmed up, the turbine was started up, and the generator was phased to the line. The load had been increased to 370 MW(e) by November 3. At about noon on November 4, a steam leak was detected on the steam lead-in line from a turbine-control valve to the high-pressure turbine; consequently preparations for load-loss tests, which had been planned for later in the day, were begun. The power level was reduced to 250 MW(e), and then both 345-kV output breakers were opened. The turbine tripped on overspeed and, in turn, caused a reactor trip. Steam was removed from the secondary system. The reactor was again brought critical about noon on November 5 in preparation for secondary-system warmup; however, at 1319 hr an unintentional ground of the vital bus scrammed the reactor. Again it was restarted, and the power output to the line was at 375 MW(e) by the end of the next day. The power level remained constant until November 10 when a scheduled loss-of-coolant-flow trip was initiated. After some minor maintenance the reactor was brought critical on November 12 for a xenon-follow test. The generator was phased to the line early the next day, and the load was increased to 400 MW(e). A load reduction to 12 MW(e) on November 14 was necessary to repair a drain line from the No. 3 turbine-control valve lead-in line; otherwise, the power level remained at 400 MW(e) until November 18 at 1720 hr when once again a grounding of the vital bus caused reactor and turbine trips. The reactor was immediately restarted, and the generator was phased and loaded to 400 MW(e). On November 20, after some simulated rod-drop tests and ramp-load-increase tests, a control rod was released to check effects of actual rod drops. Afterward the turbine was tripped manually, and a reactor scram resulted. During the maintenance shutdown that followed this scram, more low-power physics measurements were conducted.

Subsequent to the reactor-physics tests, a secondary-plant warmup was begun on November 27, and by noon on November 28 the load was at 400 MW(e). It remained at this level through December 2, when a turbine runback[9] to 110 MW(e) was effected for rod-drop tests that were part of the power-testing program. The load was back to 460 MW(e) by the end of the day. On December 8 the load was increased to 480 MW(e) with only minor problems in the steam system. The load was increased to 486 MW(e) on December 11. On December 14, fibrous material from the river water plugged the filters on the service-water intake (used for cooling various plant auxiliary equipment), and the intake had to be shifted to a deep well for a few days until the river condition cleared. The load was increased, and a maximum limit of 490 MW(e) was achieved on December 29.

## HIGHLIGHTS OF AT-POWER OPERATION

The 490-MW(e) power level[10] was maintained until late on Jan. 7, 1968, when the load was reduced to 360 MW(e) because of an ice buildup on the water-intake trash racks which decreased the water intake and thereby caused cavitation of the circulating pumps. By the following morning enough ice had been removed to allow sufficient flow, and the load was returned to 490 MW(e).

On January 12 the power load was reduced to 400 MW(e) for checkout of the turbine-stop valves. When difficulties were encountered in closing one of the valves, a decrease in reactor power was begun so that the reactor could be taken off-line. At a power level of 10 MW(e), the turbine was tripped manually. The reactor remained critical until 0815 hr on January 13 when a loose connector in the power supply to the coincidenter caused a reactor scram.

On January 14 primary-plant cooldown was begun to effect cooling-system repairs, and by late evening on January 15 the plant was in cold-shutdown condition. Primary-system maintenance was completed on January 19. Temperature-coefficient measurements were performed on January 20–21. By January 22 all essential secondary-plant maintenance was completed, and the reactor was brought critical. The turbine was rolled, the generator was phased to the line at 0149 hr on January 23, and the plant load was increased to 490 MW(e) where it stayed until February 9. A leaking feed-pump-discharge check valve[11] had to be repaired, so the power level was reduced to 260 MW(e). Routine testing of turbine-control and -stop valves was also conducted at the 260-MW(e) level, and on February 10 the plant load was increased to 491 MW(e). On February 14 at 0957 hr, the circuit breaker for the

reactor coolant-pump bus tripped open and caused the reactor to scram. The d-c control power had been removed from the coolant-pump bus-feeder circuit breaker in an attempt to locate a d-c ground. As the control power was restored, the breaker tripped open, and a reactor scram and a turbine trip followed immediately. The d-c ground, which was located on the generator antimotoring protective circuit, had been caused by a cable that was encased in ice in the underground conduit at the 345-kV switchyard. A temporary jumper was connected aboveground. The reactor was started up and the turbine rolled early in the afternoon, but difficulties with remote switching necessitated local control at the 345-kV switchyard for phasing the generator to the line. Plant load was then increased to 491 MW(e). The protective circuits in the 345-kV switchyard underground conduits were reconnected on February 16 with no difficulties.

A load runback to 350 MW(e) occurred because of a rod-bistable alarm at 0511 hr on February 20. An investigation revealed a failed rod-position coil stack, and plant load was returned to 491 MW(e). The plant remained at this level through March 1. At 2321 hr the generator[12] was unloaded in preparation for the 10-day maintenance shutdown that was scheduled to begin March 1. All maintenance was completed by March 9, and by noon on March 11 the reactor plant was at operating temperature and pressure. Early on March 12 the reactor was brought critical, the generator was phased to the line, and the load was increased to 390 MW(e). In the late afternoon a leak was found in the horizontal turbine flange in the high-pressure turbine. The reactor was shut down to repair the leak. On March 14 the reactor was started up, and steam was brought into the secondary system. By midnight the generator had been phased to the line, and the load was increased to 100 MW(e). The steam leak was again observed as before, and the plant was returned to hot shutdown. The high-pressure turbine casing was lifted for an inspection which showed that the areas adjacent to the labyrinth-gland seal covers were eroded on both ends. The eroded areas were ground out, a weld deposit was inlaid and ground down, and the areas were lapped. The high-pressure turbine was reassembled and brought into service by March 27. The generator was synchronized to the line around midnight and loaded to 60 MW(e). At 0036 hr, March 28, an unintentional actuation of the turbine low-vacuum trip signal scrammed the reactor. The reactor was restarted immediately, and the generator was put on-line soon after. At the 320-MW(e) level, the turbine-flange leak was again detected. The power was cut back to 100 MW(e) until March 29 when the plant was shut down. By March 31 the high-pressure-turbine upper cylinder had been disassembled and loaded onto a truck for shipment to the manufacturer's shop.

The turbine cylinder was returned[13] from the manufacturer's shop on April 12. By April 15 the turbine was reassembled, the reactor was critical, and plant heatup was finished. Rod-ejection tests, reactor-operator training, and rod-drop tests were completed by April 19. Secondary-system maintenance had been completed by April 18, and the turbine was rolled the next morning. A difficulty was encountered in loading the turbine when the 345-kV breaker opened. Later the high-temperature alarm annunciated from the primary-system valve-stem leakoff header. The turbine was tripped so that access could be gained to the coolant-loop areas for packing-gland adjustments. Meanwhile, a 65-V d-c multiple power-supply unit, which is used for miscellaneous data-logger printouts, failed. Although no control instrumentation malfunctioned, plant startup was delayed until a new power supply could be installed. At 2209 hr on April 19, the plant was phased to the line, and the power level was increased to 490 MW(e). The output stayed at the 490-MW(e) level until an interruption of station-service power caused an automatic trip at 1300 hr on April 27. The station remained on emergency power for about 25 min while the 115-kV power was restored. One of the rod-position-indicating solenoid stacks also had to be replaced before the reactor could be started. By 2213 hr on the 27th, the plant had been returned to the line and loaded to 490 MW(e) where it remained until May 3.

The load[14] had to be cut back to 300 MW(e) on May 3 to take coolant loop No. 2 out of service so that a weld leak at the flow orifice of the No. 2 feedwater-supply line could be repaired. The next morning the load was returned to 490 MW(e) but had to be reduced again to 300 MW(e) for an outage of the 345-kV transmission line.

During a routine check of power-range set points on May 4, a false overpower trip signal on two of the power channels tripped the reactor at 0855 hr. The plant was brought back on-line by 1316 hr, and the load was increased to 300 MW(e). The power level remained at 300 MW(e) until the following day so that the 345-kV line could be restored to service. The power was then increased to 490 MW(e). On May 16, with the power level at 490 MW(e), a partial collapse of the circulating-water weir caused the water level in the discharge tunnel to drop about 6 in. Stop logs were inserted into the tunnel to prevent a further decrease in

water level in case the weir completely collapsed. On May 19 the plant load was reduced to 300 MW(e) for about 17 hr for a maintenance outage of the 345-kV line. The load was then returned to the 492- to 496-MW(e) level where it remained through June 5.

On June 5 a normal shutdown[15] was effected to allow physics testing and secondary-system inspections. On June 9, with the reactor critical, the operator noticed that one of the control-rod subgroups had dropped, and he scrammed the reactor. The movable gripper coil for the subgroup had been inadvertently deenergized while the lubricating oil was being changed in the slave-cycle gearbox. The reactor was restarted, and at 2226 hr the generator was phased and the load was increased to 486 MW(e).

On June 10 a loss-of-coolant-flow signal caused a reactor scram at 1247 hr. A malfunction in the flow transmitter had caused an erroneous signal. The plant was shut down to permit work on the flow transmitter inside the coolant-loop area. Repairs were completed, and by 1857 hr the power output was 365 MW(e). On the afternoon of June 12, plant load was again increased to 490-MW(e). During the weekly containment inspection on June 14, a pinhole leak was found in the body of loop No. 2 feed-line check valve. The power was reduced to 365 MW(e) so that loop No. 2 could be isolated and the leak could be repaired. However, the plant was shut down at 0830 hr on June 16, prior to returning the loop to service. By 1206 hr the same day, the reactor was back on line and loaded to around 490 MW(e) where it remained until July 4. By July 4 at 0030 hr, the load was reduced[16] to 300 MW(e) to allow a scheduled 345-kV transmission-line outage. The line outage extended through the late evening of July 6. However, the plant was shut down at 1617 hr on July 5 for repairs on the steam-generator feed pump. The mechanical seals on the pump were replaced, and the casing drain plugs on both feed pumps were seal welded. At 2130 hr on July 6, the plant was back on line, and the load was increased to 486 MW(e). On July 16, difficulties were encountered when an addition of boron to the coolant system was attempted. The primary-water transfer pumps were gas bound because the makeup stop valve was leaking and was allowing volume-control-tank gases to leak back into the primary system. The back leakage was stopped, and the primary water-transfer system was vented so that the primary water-transfer pumps could operate normally. The load remained at the 470- to 478-MW(e) level until 1800 hr on July 19 when preparations were begun for taking the reactor off-line. The generator was unloaded at 2327 hr. The plant shutdown was scheduled so that the No. 1 turbine-control valve could be opened and inspected. A new gasket was installed in the feed-line check valve to the No. 1 steam generator, and the loop was returned to service by 2100 hr on July 21. When the turbine-control valve was reassembled, the reactor was brought critical, and steam was returned to the secondary system. On July 22 the generator was phased to the line at 0236 hr and the power was increased to a 470- to 480-MW(e) level where it remained until August 2.

On August 2 the load[17] was reduced to 400 MW(e) for a routine turbine-control and -stop valve test. At 1852 hr, during a test on the right-hand stop valve, both stop valves closed, an action that shut down the turbine and tripped the reactor. A check valve in the oil-discharge line from one of the servomotors had become stuck in the open position; the resulting discharge of oil through one of the solenoid dump valves caused the stop valves to close. Steam was removed from the secondary system so the stop valve could be disassembled. Primary coolant loop No. 2 was isolated and cooled down so that adapter seals could be installed and welded on the cover flange of the feed-line check valve.

On August 5 at 0925 hr, the reactor was brought critical; by 1141 hr the generator was phased, and the plant load was increased to 476 MW(e). During reprogramming of the turbine-control valves, the reactor was again scrammed from 400 MW(e) by a closure of both valves that caused the turbine and the reactor to trip. A piece of weld slag was found under the seat of the solenoid dump valve in the automatic-stop oil line from the right-hand stop valve, and the relief valve in the line had the adjusting nut backed off. Two of the four turbine-control valves were removed and inspected in detail. On August 13 the generator was phased to the line at 0840 hr and the load level brought to 476 MW(e). The power load varied between 400 and 475 MW(e) until 1128 hr on August 22 when the feedwater regulating valve of the No. 3 loop closed. This caused a steam-flow–feedwater-flow mismatch coincidentally with a low steam-generator water level and scrammed the reactor. A loss of control air had caused the valve to close. The generator was phased back to the line at 1316 hr, and the output load was returned to about 400 MW(e).

On August 29 a plant shutdown was initiated for the purpose of inspection and repair of the turbine-control and -stop valves. The plant was returned[18] to the line at 1905 hr on September 5. The load was

increased to 400 MW(e), and tests showed that the turbine-control and -stop valves were operating correctly. Plant load was then increased to the 490-MW(e) level. On September 21 the plant had to be taken off-line so that a solenoid dump valve in the automatic-stop oil line for one of the turbine-control valves could be replaced; otherwise the plant remained between 480 and 490 MW(e) throughout the month.

In October some problems[19] were still encountered in equalizing the pressure across the turbine-stop valves during routine testing; but control-valve seating was facilitated by tapping with a hammer, and the control-valve seating allowed the stop valve to be closed. On October 6 one of the charging pumps failed in service. An alternate letdown mode of operation was invoked with another charging pump in standby reserve. Repairs to the failed charging pump were completed, and normal operation was resumed on October 24. Plant load level remained at a nominal 490 MW(e) during October with an exception of one load runback on October 13 to 445 MW(e) which was caused by control-rod slippage. The rod slipped, during normal reactor operation, from 307 steps to 265 steps, actuating a load runback from a neutron-flux rate-of-change indicator.

On November 18 an automatic reactor trip[20] from ~494 MW(e) occurred without apparent cause. The annunciator panel in the reactor control room indicated a turbine trip, and the turbine annunciator indicated a reactor trip. A computer printout of events showed that a scram breaker opened to initiate the trip. All shutdown systems functioned normally to bring the plant to a hot-standby condition without action by the operators. All systems and components passed inspection and checkout. At 1405 hr the same day, the reactor was returned to criticality without problem. The plant was phased to the line at 1511 hr; however, problems with the No. 2 turbine-control valve prevented operations. The plant was taken off-line at 1543 hr, and the turbine was shut down until the valves could be repaired. At 2356 hr on November 18, the plant was returned to the line, and normal full-power operations were resumed. Difficulties with the turbine-control valve had been caused by metal filings and chips binding the pressure-feedback sleeve of the control oil pilot, so that no feedback was generated and the valve behaved erratically. The sleeve was cleaned and deburred, the servomotor was reassembled and adjusted, and the control valve was satisfactorily tested. On November 27 the plant was shut down for maintenance and reactor-physics testing and was returned to service on December 2.

With the power level at a nominal 490 MW(e), an automatic load runback[21] was initiated on December 7 by a failure of one of the vital-bus inverters. The vital bus was transferred to the backup power supply until the transformer in the inverter was replaced. On December 9 another plant trip occurred without apparent cause, and again the plant was brought to hot standby automatically. The computer again indicated that a trip breaker opened to initiate the trip action. (Subsequent to the November 18 trip from an unknown cause, detailed inspections of the trip breakers at the end of November had shown the breakers to be in excellent condition. Also, tests had shown all plant trip circuits to be performing properly.) After a checkout of systems and components revealed no problems, the reactor was brought critical, and the plant was phased to the line at 1605 hr on December 9. The plant was at full power 3½ hr later.

Wiring checks during the November 27–December 2 shutdown showed that the coil leads for the time-delay relays in the undervoltage trip circuit for the coolant pump were disconnected. An operating order was issued to ensure that the breakers would be tripped in the event of a power loss before the next scheduled shutdown when the condition could be remedied.

On December 17 the reactor was scrammed again from nominal full power and from no apparent cause. All automatic actions were reviewed for correct operation. Instrumentation and circuitry functioned normally. Generator and switchgear relay targets and switching actions were checked and verified as being required by the trip. In an attempt to pinpoint the cause of the trips from an unknown source, the two plant trip breakers were exchanged even though they had previously been cleared by both mechanical and electrical checks. After the trip-breaker exchange and several hours of wiring checks, plant startup was begun at 0000 hr on December 18. During the approach to criticality, problems were encountered with the control circuitry for the control-rod groups, and startup was delayed. Criticality was achieved at 0343 hr, and full power was reached at 0800 hr.

Problems with the control rods occurred when bank A was selected and withdrawal was begun. It was immediately discovered that bank B was being withdrawn simultaneously even though it was not selected. Both banks indicated inward motion when the rods were driven in. The bank-selector switch was turned from A to B, and the action was repeated. The rods were then fully inserted. A checkout of the rod-control-system equipment showed that one of the

memory relays in the bank-A overlap assembly was manually rotated one step. Further checking showed that the command relay for bank-B operation was mechanically held in the energized position, so that, when bank A was selected, the command relays for both banks became energized and rods in both banks operated on a stepping signal. The relay was removed, repaired, and reinstalled, and normal response of both banks was verified before a second approach to criticality was begun.

With the reactor at nominal full power,[21] another plant trip occurred on December 25. This was the fourth plant trip without apparent cause in less than 2 months. Again the computer printout indicated that the same trip breaker had opened to initiate the trip action. All automatic actions were consistent with the past automatic-trip occurrences, with the exception of the main-generator disconnect switch that failed and remained closed. A shear pin in the generator drive-unit gearbox had sheared and was replaced. However, an attempt to open the disconnect switch manually caused the pin to shear again. The drive linkage was disconnected, and attempts were made to open the individual switch blades. Two blades were opened, but efforts to open the third blade caused another pin to shear. Because of the time and manpower requirements to completely disassemble the switch, the blades were closed, and the switch was disconnected temporarily.

Since the breakers had been previously interchanged, it was tentatively concluded that the trip-coil circuit had initiated the trip. A wire-by-wire check of associated circuitry gave no conclusive evidence. The reactor was brought critical at 0202 hr on December 26 and was at full power by 0630 hr. Late on December 27 the plant was shut down for a comprehensive investigation of the plant protective instrumentation in an attempt to find the cause of the unexplained plant trips. The plant was returned to service on the morning of Dec. 30, 1968, to the base load of ~497 MW(e).

On Jan. 6, 1969, an automatic load runback[22] to 385 MW(e) was initiated at 0645 hr because of a failed vital-bus-power-supply inverter. The vital bus was transferred to the backup power supply. The plant load was back at full power by 0800 hr. A failed capacitor in the inverter had caused the inverter to fail. A spare transformer was installed in the inverter, and the bus was returned to its normal power supply. At 0904 hr on January 8, another automatic trip from an unknown cause occurred. Again the same breaker had opened to initiate the action. A newly installed relay in the breaker trip circuitry indicated that the trip signal was initiated from one of the permissives that operated between 10 and 84% of full power. Since all circuits in this group had been functionally and visually checked wire-by-wire during the last shutdown, it was assumed that some component within one of the circuits was producing a spurious signal. Furthermore, it was deduced that the autostop oil-pressure trip was the only circuit in the group that could initiate a direct trip of the breaker without an alarm or annunciation. The double-bulb mercoid switch in the oil-pressure trip was replaced, and the old switch was connected to an alarm circuit for further evaluation. The trip signals above the 84% full-power level and the trip signals from the autostop oil system and the stop-valve closure circuits were monitored. By 1337 hr on January 8, the reactor was again critical, and full power was achieved by 1620 hr. During repair of a leak in the autostop oil system, one of the turbine-stop valves was unintentionally closed, and a gradually decreasing load transient, observed from the reactor control room, caused the shift supervisor to order the reactor manually scrammed at 1735 hr the same day. By 2320 hr the plant was back at full power.

The plant was at the base load[23] of 497 MW(e) throughout February 1969 with the exceptions of cutbacks to 300 and 375 MW(e) for transmission-line work. One of the two incoming 115-kV station-service power lines was downed during a severe snowstorm on February 9; however, the power loss did not stop plant operations, and the line was restored quickly.

During the power reductions for the transmission-line work, the circulating-water side of both condensers was backwashed to remove some of the debris that had collected.

On February 24 at 1220 hr, a full-power xenon oscillation test began and continued through the end of the month. The test required that the rod positions, at various bank heights, he held constant and that $T_{av.}$ be controlled by either small load changes or borations and dilutions.

Approval[24] was received on March 11 to increase power from a nominal 1473 MW(t) [~500 MW(e)] to a nominal 1825 MW(t) [~600 MW(e)]. The plant was near the base load of 497 MW(e) until March 13, and then the power level was increased stepwise to the 600-MW(e) level. The problems encountered in increasing the power to the 600-MW(e) level were in the feed–condensate system and the moisture-separator–reheater-drain systems. A feed pump was tripped at 550 MW(e) by a small switching transient that actuated the low-suction-pressure trip. It could not be restarted

because of a 27-psi reset margin, which was later reduced to 5 psi. An erratic level-control valve in the discharge line of the heater-drain pump caused spikes that resulted in momentary pressures for the feed pumps which were as low as 220 psi.

The plant was base loaded at ~600 MW(e) until April 10 when power output[25] was reduced to 575 MW(e) in preparation for a shutdown on April 11. The shutdown had been scheduled for turbine-valve modifications and primary-system maintenance and testing. The reactor was maintained in the critical state until 0618 hr on April 12 when it was manually scrammed. All rods responded to the scram signal except No. 28, and all rod-bottom signals annunciated except No. 28. A position readout on the digital voltmeter showed 307 to 308 steps withdrawn. Resistance readings on the position-indicating coil eliminated the possibility of false-position readout. When the trip breakers were closed, the rod moved out about three steps, and the digital voltmeter indicated normal movement. The trip breakers were opened and reclosed, and the rod moved outward a couple of steps but would not move inward. Finally, after the second outward movement, the rod responded to an inward signal and was moved in several steps. A trip signal was then given, and this rod responded normally. The rod was exercised, and a detailed program was conducted to verify the operability of the rod. Further exercises were performed without difficulty.

During the shutdown the main generator was completely overhauled; the stator-coil wedges, spacers, and insulation were disassembled and repaired. After completion of primary-system maintenance, preparations were begun for plant startup. On May 10 the turbine generator[26] was rolled at 2031 hr, but was shut down immediately because of contact noises that were detected within the generator, exciter, and permanent-magnet generator. Subsequent investigations showed that the shroud and blades of the hydrogen fan and the exciter cooling fan were rubbing and that a bolt in the permanent-magnet assembly was loose and was scraping the rotor of the pilot alternator. The problems with the hydrogen fan and exciter cooling fan were eliminated by increasing shroud clearances. The pilot-alternator rotor had been damaged by the loose bolt and had to be removed and repaired. The functions of the alternator were assumed by three temporary single-phase 115-kVA transformers until the rotor was returned to service. The generator was phased to the line at 1805 hr on May 11, and plant load was maintained on a programmed schedule until June 6.

For the remainder of 1969, a programmed schedule was adopted so that the full 1825-MW(t) capability would be available over the 1969–1970 winter months. The schedule called for the plant to be loaded at 80% of full power [~490 MW(e)] from Monday through Friday evening and a reduction to ~150 MW(e) at all other times. This programmed schedule[27] was maintained from May to December 1969 when the power level was returned to ~600 MW(e).

On May 3 at 1015 hr, one of the boron-recovery-evaporator distillate test tanks[26,27] was pressurized until it cracked at the disked head seam. An improper valve lineup had caused the pressurization and was immediately corrected. On May 6 between 1500 and 1700 hr, about 500 gal of waste liquid, concentrated to 12% boric acid, was discharged from the first-stage bottoms of the boron-recovery evaporator to the evaporator floor area and by floor drains to the aerated drain tanks because of a broken pipe nipple. When the discharge was discovered, the affected lines were shut off, the evaporator was cooled down, and decontamination efforts were begun. Neither of these occurrences affected plant operations.

On June 6 at 2039 hr, the plant[28] was shut down to correct hydrogen-leakage problems in the generator and to make repairs on one of the feed-line check valves. The plant remained out of service until June 9 when the load was established at 485 MW(e). A mechanical failure of one of the feedwater flow-control valves caused a manual trip from 485 MW(e) to be initiated at 0315 hr on June 10. The valve plug had broken off the stem, and the plug had dropped open, allowing full feedwater flow to the steam generator. The valve was repaired, and the plant was back in service by 2059 hr on the same day. On June 12 at 1131 hr, a load runback from ~480 MW(e) to 355 MW(e) was caused by a spurious signal from a power-range channel. Normal plant conditions were verified, the plant load was returned to normal, and the power-range channel was removed from service for further inspection. The power level remained on the programmed schedule until July 15. A loss[29] of station-service power on July 15 caused a plant trip from 480 MW(e) with the plant remaining on emergency power for 9 min. During a switching routine a procedural error had been made which allowed both service power lines to trip at once. A check of systems and components subsequent to restoration of power revealed that the mechanical seals on the No. 4 reactor-coolant pump had failed in the outage. The No. 4 loop was isolated, the reactor was brought critical, and, at

2118 hr on the same day, the generator was phased to the line. The plant load was maintained at a nominal 440 MW(e) until 2248 hr on July 18 when a plant shutdown was begun for repairs to the No. 4 coolant pump. The loop was cooled down and depressurized, and the rest of the primary system was maintained in hot standby. All three mechanical seals in the coolant pump were replaced and tested, and the pump was returned to service. The No. 4 coolant loop was returned to service on July 21, and the plant was back on-line at 0823 hr.

At 0313 hr on August 2, the reactor[30] was tripped from 440 MW(e) by a lightning strike in the vicinity of the electrical relaying equipment. The plant was automatically brought to the hot-shutdown condition. The containment system and all other systems were checked out before the plant was returned to service at 1020 hr. On August 18 a manual plant trip was initiated at 0609 hr when two control rods dropped. All wiring, contacts, relays, and slave-cycler cam-operated contacts checked out normally. When the checks were finished, current traces were made with the Visicorder as one of the dropped rods was being exercised. Also, voltage traces of the movable and stationary gripper coils were made. The rods functioned normally during exercises, and the rod-control system was returned to service at 1422 hr.

At 0640 hr on August 30, the plant was removed from the line to replace two failed resistance temperature detectors (RTD) in the cooling-system $\Delta T$ instrumentation and remained shut down until September 1. On August 29 at 2023 hr, the plant was at 520 MW(e) when the loop $\Delta T$ readout for loop No. 3 began oscillating between +60 and −20°F. The problem was traced to a failed RTD in the loop. A plant-load decrease was initiated at 0319 hr on August 30 in preparation for cooldown of loop Nos. 3 and 4. Repairs to the temperature-detecting instrumentation were completed, and the loops were returned to service on the afternoon of August 31. During a critical approach that afternoon, one of the control rods stuck at 200 steps. Attempts to free it resulted in a position change to 87 steps, and so a plant cooldown was initiated. At 375°F a periodic attempt to induce rod motion resulted in moving the rod inward by five steps. A trip breaker was then opened, and the rod responded normally. The rod checked out normally when it was exercised. On September 1 the reactor was brought critical at 1215 hr, and the generator was returned to service at 1431 hr.

Plant load[31] was increased from 400 to 480 MW(e) on September 2 where it remained until 0040 hr on September 6. At this time the plant was shut down for repair of a feedwater heater and replacement of an RTD in coolant loop No. 3. The plant was returned to service on September 7, and, during the load increase, two of the control rods were observed to slip about 100 steps into the core. The power level was maintained at 316 MW(e), and the two rods were placed in the disconnect mode for tests. All checks and tests indicated normal response of the rods, and plant operations continued. By September 30 the plant load was at 580 MW(e).

The plant[32] was removed from service at 2203 hr on October 6 because oil pressure in the turbine governor was decreasing. A worn guide block in the pump impeller sleeve was replaced, and the dashpot in the orifice relief valve was modified and smoothed out. The plant was returned to service at 1055 hr on September 7. Problems in the feedwater system necessitated a plant shutdown for the weekend beginning at 0025 hr on October 18. Twenty-seven leaky tubes in one of the feedwater heaters were plugged, the mechanical seals on one of the feedwater pumps were replaced, and the slave-cycler clutch assembly for one of the control-rod subgroups was replaced. The plant was returned to the line on October 20 at 0222 hr, and the load was increased to 600 MW(e) at 0800 hr on September 21 after a required 24-hr delay. The plant remained base loaded at a nominal 600 MW(e) until an automatic trip[33] occurred on November 11 at 2317 hr. A steam-flow–feedwater-flow mismatch signal concurrent with low-level signals from all four steam generators, which initiated the trip, had been caused by a momentary closure of the turbine-control valves while some adjustments were being made to some oil-system components. The plant was returned to service on November 12, and the load was at 490 MW(e) by 0640 hr. At 0818 hr an automatic trip from 490 MW(e) was again triggered by steam-flow–feedwater-flow mismatch and low-level signals from three of the steam generators. Once again adjustments were being made in the oil system of the turbine when the control valves closed momentarily. The reactor was maintained at critical, and the plant was placed in hot standby while an investigation of the turbine-governor oil pressure was conducted. The impeller sleeve for the turbine-governor oil pump was replaced, and the turbine was checked out and returned to service. The plant was returned to the line at 1619 hr on November 13.

Operation at a nominal 600 MW(e) was interrupted on November 18 by an automatic trip at 0843 hr. A primary electrical protection relay annunciated as the

first-out trip. The plant was maintained in hot standby while the switchyard and relay terminals were examined. Moisture was found in the terminal box at the neutralizing transformer in the 345-kV switchyard. The general consensus was that a voltage track had occurred between the power and relay terminals and caused the trip. The terminal box was dried, and all incoming conduits were sealed. The reactor was brought critical, and the plant was returned to the line at 1202 hr the same day. On November 28 at 0308 hr, immediately after the plant load was reduced from 500 to 400 MW(e) in preparation for turbine-valve tests, a rods-out-of-step alarm was annunciated. During the rods-realignment actions, two more rods in the subgroup were dropped, whereupon a manual plant trip was initiated. Investigation of rod-control circuitry revealed a faulty time-delay relay in the holding-power circuit for the two rods. The relay was replaced, and no further problems were encountered with these rods. The checkout of the original dropped rod indicated that it was lagging behind the bank and did not respond normally to movement signals. Later investigations revealed a faulty smoothing capacitor in the position-indicating circuit. The capacitor was replaced, and position indication returned to normal.

During the turbine-valve testing on November 28, one of the turbine-stop valves was closed and failed to reopen. Apparent cause of valve failure was excessive differential pressure across the stop valve because of leakage through the control valves. Adjustments to the pressure-control relief valves in the stop-valve servomotors restored the autostop oil pressure, and no further problems were encountered. The plant was returned to service on November 30 but with all four reheaters and steam to one of the feedwater heaters out of service. These components[34] were returned to service on December 4. The power level was then increased to a nominal 600 MW(e) until December 20 when an automatic load runback was experienced at 2004 hr. A malfunction in the circuitry of one of the power-range instrumentation channels caused a rod-drop rod-stop alarm that resulted in a turbine-load runback. Since all control-rod positions and bottom bistable indications were normal, the load-runback feature was defeated, and the turbine load was restored to 600 MW(e). The faulty channel drawer was replaced, and the load-runback feature was returned to service. At the end of 1969 the plant was still base loaded at a nominal 600 MW(e). The plant remained[35,36] at this level until Feb. 7, 1970, when the end of life for Core I was reached. Coastdown[36] began and continued at a rate of about 2 MW(e) per day until February 19 when the power level was reduced to 380 MW(e) in preparation for isolation of coolant loop No. 4. Increasing tube leakage from the primary to the secondary side of the steam generator necessitated the loop isolation. After the loop was removed from service, the load level was increased to a nominal 455 MW(e) where it remained[36,37] through March 21.

During the last week of January 1970, the radiation monitor for the main condenser air-ejector effluent showed a gradual increase[36] to the extent that a primary-to-secondary leak was suspected. Gross activities of the secondary side of the steam generator are radiochemically determined on a weekly basis. These determinations confirmed the presence of activity in the secondary system, and a comprehensive radiochemical-analysis program was effected to determine which of the four steam generators were leaking. When the No. 4 loop was isolated on February 19, the air-ejector activity rapidly returned to normal. The leaking tubes were scheduled for plugging at the imminent refueling outage.

On March 21 at 1024 hr, the plant[37] was shut down for repair of the motor-operated valve in the steam-generator feed-pump discharge. At 1523 hr, with the steam plant shut down and the reactor just critical, a control-rod slippage was noted. The control rods were being adjusted outward to stabilize the reactor-coolant average temperature. After the final rod positions were verified, two of the four control-rod banks were fully inserted and scrammed the reactor. A faulty time-delay-relay microswitch in the rod-control circuitry was replaced, and the time-delay relay was reset. Normal operation of the rods was verified, and the reactor was returned to critical. At 0038 hr on March 22, with the reactor just critical and procedures in progress to roll the turbine, a steam-flow–feedwater-flow mismatch signal from the No. 3 steam generator scrammed the reactor. An erroneous steam-dump signal, which was actuated during verification of the $T_{av.}$ signal from loop No. 3, resulted in the scram action from the steam generator. The $T_{av.}$ circuitry was restored to normal, and at 0105 hr the reactor was again critical. On March 25 the plant was shut down for about 5 hr to repair a leak in the No. 3 steam-generator feedwater drain line. The body of the feedwater control valve was also replaced since a small leak had developed there. The reactor was returned to the nominal 450-MW(e) level where it stayed until April 2. At 1253 hr an automatic load runback[38] was caused by control-rod slippage from 317 steps to 241 steps during weekly rod exercises. Plant conditions were verified after the power stabilized at about

350 MW(e), and, as attempts were being made to return the slipped rods to their normal position, one of the rods dropped to the full-in position. Although no significant effects were observed, the power level was reduced to 300 MW(e). All rods in the bank were placed in the disconnect mode, and an attempt was made to retrieve the full-in rod. During the first step out for this rod, all the other rods slipped into the core. The reactor was scrammed from 300 MW(e), and the plant was automatically placed in hot-standby condition. Investigation into the rod-control circuitry revealed a faulty contact in the relay that controls the sequencing of the stationary gripper coil. The faulty contact resulted in a deenergized lift coil that allowed the rods to slip. The plant was returned to service at 1630 hr on April 2.

A steam leak in the sensing line of the No. 3 feedwater-flow-control orifice necessitated a steam-plant shutdown for repairs at 0137 hr on April 14, but the reactor was maintained at critical. At 0411 hr an automatic reactor trip occurred. A noise spike from one of the power-range reactor-control channels in conjunction with a checkout of the overpower reactor-trip set point for a second power-range channel satisfied the two-out-of-four trip logic and shut down the reactor to the hot-standby, subcritical condition. The circuitry for the two channels was thoroughly checked with no problems or discrepancies being evidenced, and by 0650 hr, when the steam leak was repaired, the plant was returned to the line. At 2309 hr on April 17, the plant was removed from service at 300 MW(e) by an induced electrical trip for the refueling outage. The purpose of the trip from power was for an evaluation of the resultant turbine overspeed. The reactor remained shut down[38-40] from April 17 through June 26.

Refueling was accomplished[39] without difficulty, and the upper internals package was installed by June 3. On June 20 the reactor[40,44] was brought critical by boron dilution at 1340 hr for zero-power physics testing. Coolant loop No. 3 was isolated, depressurized, and cooled. On June 21 a working party in the containment area reported a noise in the vicinity of the pressurizer, and a coincident alarm from the data logger indicated a low pressurizer pressure. The closed-circuit television camera, when focused on the top of the pressurizer, showed steam blowing from the manway on top of the pressurizer. The reactor was scrammed, and a plant cooldown was begun immediately. Inspection revealed that a gasket of the wrong material had been used and that a ½-in.-wide section had blown out. The proper gasket was installed, a hydrostatic test was performed, the plant was heated and pressurized, and the physics testing program was continued. The No. 3 coolant loop was returned to service on June 24. On June 26 the turbine was rolled and vibration tested, and the generator was phased to the line at 1831 hr; thus the refueling and maintenance shutdown ended. The plant was removed from the line on June 27 for additional turbine balancing and was loaded to 48 MW(e) on June 28 for an adjustment of the impeller-supply orifice on the turbine governor and for a check of the overspeed-trip set point. On June 29 the generator was phased to the line at 0140 hr and was loaded to 300 MW(e) for 50%-power equilibrium-xenon testing. The plant load was temporarily increased to 360 MW(e) for vibration measurements on the control valve of the No. 1 turbine and was then returned to the 300-MW(e) level in preparation for 50%-load-rejection tests, which were completed July 2. That same day[41] the plant was returned to service and to full power where it stayed until July 19. A low-vacuum condition in the main condenser prompted the removal of the plant from the line on July 19. Later inspections showed that one of the gland-seal steam strainers was plugged. On July 20, after all strainers were removed, cleaned, inspected, and reinstalled, the plant was returned to service at full power until August 7. The power level[42] was reduced to 400 MW(e) on August 7 for weekly turbine-valve tests, but an improper functioning of one of the stop valves resulted in plant shutdown at 0306 hr on August 8. The shaft of the stop valve was replaced, and the valve operated properly. The plant was returned to full power at 0807 hr on August 10. During a severe electrical storm on August 17, an automatic plant trip occurred at 1529 hr. The plant was maintained in hot shutdown during investigation of the trip cause. One of the terminal boxes in the 345-kV switchyard evidenced voltage tracks between the telephone and pilot-wire terminals, and the pilot-wire fuses were blown. These conditions were corrected, and, since all other conditions were normal, the plant was placed back on-line and was at full power by 2200 hr.

At 1518 hr on August 19, a two-man work party reported to the control room that they had extinguished a small, localized fire at the juncture of the loop No. 4 coolant-pump suction piping and the pump volute. At 1540 hr the plant load was reduced to 65% of full power, loop No. 4 was removed from service, and a working party entered the containment to assess the extent and source of the problem. Oil was leaking from the upper oil reservoir of the coolant-pump motor onto the pump-bowl insulation and slowly

running[42-44] down the bowl until it reached an opening in the pipe insulation, contacted the hot pipe, and then flashed. The actual burned area was about 6 in. in diameter and was surrounded by an area of soot and char about 18 in. in diameter. The area was cleaned, and the leak was apparently repaired. The plant load level was maintained at <75% of full power with loop No. 4 out of service until 1126 hr on August 20. The No. 4 coolant pump was started, and the loop was returned to service. By 1615 hr on August 20, the plant was back at full power. On August 26 the power was again reduced to 65% of full power to permit the isolation of coolant loop No. 4. The insulation was removed from about half the pump bowl and from the hot-leg pipe. A slight discoloration was noted in the pump bowl in addition to the oil-soaked and charred insulation. All residue and discoloration were removed by cleaning with a wire brush, and the bowl and piping were reinsulated. The loop was returned to service at 1317 hr on August 30, and the plant was at the nominal 600-MW(e) level by 2139 hr.

On September 3 the plant[43] was removed from service at 2025 hr to repair a leaking valve packing gland. At that time the packing was also adjusted on several primary-system plant valves in the containment-loop areas. The plant was returned to service at 0809 hr on September 4. A cold shutdown was effected at 0732 hr on September 10 to repack the hot-leg stop valves on coolant loops Nos. 1 to 3. Also, the packing was replaced above the lantern rings in the bypass valves for all four loops. At 0032 hr on September 13, the plant was again at full power.

An automatic relay action[43,44] at 1803 hr on September 18 isolated one of the station-service transformers and left the plant on a two-line, one-service-transformer configuration. The turbine load was reduced to about 470 MW(e) to stabilize the amperage and winding temperature on the other transformer. Eventually the load was reduced to 50%, but, after observation and evaluation, it was returned to about 85%. A substitute transformer was located at a nearby plant, the failed transformer was removed for repairs, and the substitute was installed in less than a week with no plant downtime. The plant was operating at the nominal base load of 600 MW(e) at the end of September 1970.

## MAINTENANCE AND REPAIRS

The items listed below, taken directly from the referenced reports, will give the reader an idea of the minor problems which can be encountered during routine operations and which do affect plant availability.

### Plant Systems

#### July 1967 (Ref. 4)

A boric acid pump was overhauled.
Antireverse rotation pawls were installed on the reactor-coolant-pump motors.
Mechanical seals were replaced on one of the residual-heat-removal pumps.
Check valves were installed in the discharge piping of both test-tank effluent pumps.
Both condensate pumps were repacked.
Calibrated hydraulic snubbers were installed on the neutron-shield tank for earthquake protection.

#### August 1967 (Ref. 5)

Defective motors in the motor-operated disconnects in the 345-kV switchyard were replaced.
Several leaks in the main-generator hydrogen system were repaired.
The originally installed 50-A starter on one of the diesel generators was replaced with an 80-A starter.
A defective control-rod position-indicating coil stack was replaced with a spare.
Excessive leaks in both steam-generator feed-pump recirculation lines were repaired.
Excessive leaks in seal-water-flow orifice plugs were repaired.
Torque limit switches on all feedwater-system motor-operated valves were reset.
A second boric acid pump was replaced with one of new design (see list for July 1967).
Glands on both charging pumps were replaced.
Leaks in discharge piping of the purification pump were repaired.
Leaks in the flanges of the reheater steam-supply piping were repaired.
Failed impellers in both waste-gas blowers were replaced.
The 1½-in. suction piping for the waste-liquid-transfer pump was replaced with 3-in. piping.
Isolation valves were installed on the steam-generator blowdown lines in the containment.
Leaking pipe caps on the casing vents for both charging pumps were repaired.
Failed seals on the recycle pump in the boron-recovery system were replaced.
A caustic pump in the water-treatment plant was overhauled and reconditioned.
A voltage regulator in the control-rod power-supply cabinet was overhauled and repaired.
Failed packing in a heater drain pump was replaced.
A leak in the oil system for one of the reactor-coolant pumps was repaired.
New seats were cut and new disks were installed in all three of the pressurizer safety valves.
A failed head-joint gasket in the nonregenerative heat exchanger was replaced.
A bellows leak in the relief valve of one of the charging pumps was repaired.

A steam-packing leak in one of the reheater supply valves was repaired.
A leaking steam-dump valve was relapped.
Hydraulic operators on both pressurizer spray valves were replaced with air operators.

**September 1967 (Ref. 6)**

New in-core thermocouples were installed.
The seal in one of the reactor-coolant pumps was replaced, and the motor was balanced.
Leaking valves and failed socket welds in the coolant-letdown-monitor piping were repaired.
Failed packing in the isolation valves for one of the coolant loops was replaced.
Failed packing in both pressurizer spray valves was replaced.
Failed strainer cover on service-water pump was replaced.
Two leaking main-steam-line safety valves were relapped.
Closure solenoids were installed on the four main-steam-line trip valves.
The seal-oil regulator for the hydrogen system was overhauled.
Pressurizer safety valves were relapped on three different occasions.
Weld failures in the recirculation lines for both steam-generator feed pumps were repaired.

**October 1967 (Ref. 7)**

The vent valves were removed from the drains on the main steam lines for the turbine, and the valves were plug welded.
Three of the feedwater-heater relief valves were lapped, tested, and reinstalled.
The gate valve in a reheater steam-supply line was relapped and a new bonnet gasket was installed.
Two aerated drains-tank pumps were repacked.
Bonnet gaskets on two secondary safety valves were renewed.
Leaks were repaired on the following: a vent line to two feedwater heaters, a reheater drain and reheater drain tank; a 0.25-in. plug on a check valve in a feedwater-heater steam-extraction line; and a relief valve in the letdown line.
The control valves for the reheater drain tank were overhauled.
A leak in the main-steam line piping between a turbine-control valve and steam chest was repaired.
Two heater drain pumps were repacked.
All four turbine-control valves were disassembled for inspection and repair.
A crack in the body of the steam-generator feed-pump discharge was ground and welded.

**November 1967 (Ref. 8)**

The bonnet gasket in the main-steam-line drain-header trip valve was replaced, and the packing was lubricated.
The damaged rubber boot in the temperature-control valve for the nonregenerative heat exchanger was replaced.
One of the feedwater-regulating bypass valves was dismantled, metal chips in the seat were removed, the seat was repaired, and the valve was reassembled.
A frozen piston was replaced in one of the water-treatment acid pumps.
New stems were installed in two of the reheater-control valves.
A feedwater-heater relief valve was relapped.
Four new vital-bus sections and inverters were installed.
Mechanical seals on one of the steam-generator feed pumps were replaced.
Five pressurizer capillary-tube valves were replaced.
A new feedwater-regulating valve was installed.
The cracked yokes of two of the feedwater-station motor-operated isolation valves were welded.
Motor operators were installed on the residual-heat-removal–charging-system cross-connect valves.
Both flange gaskets were replaced on the level-control valve for the reheater drain tank.
The upper motor bearing on one of the main condensate pumps was replaced.

**December 1967 (Ref. 9)**

New impellers were installed on the waste-liquid-transfer pumps.
Vibration dampers were installed on the main steam lead-in lines for two of the turbine-control valves.
A leaking bonnet was repaired on a feedwater-regulating bypass for an isolation valve.
A leaking capillary line in the containment vessel was welded.
The packing on both charging pumps was adjusted.
A new solenoid valve was installed in the primary water-transfer makeup line to the reactor-coolant-pump vapor-seal head tank.

**January 1968 (Ref. 10)**

Two pressurizer safety valves were relapped.
Both turbine-stop trip valves were overhauled.
The coupling-end motor bearing was replaced on one of the steam-generator feed pumps.
The seat on the steam-driven feed-pump control valve was relapped, and the manual isolation valve was repacked.
One of the secondary safety valves was relapped.
A new isolation valve was installed upstream of a feedwater-regulating bypass valve.
Two new valves were installed in the coolant-pump seal-water return line.
A pinhole leak in a feedwater check valve in the containment was repaired.
Bottom blowdown isolation valves were installed on one of the steam generators.

**February 1968 (Ref. 11)**

A pinhole leak in the body of one of the discharge check valves for a heater drain pump was repaired.
A leak on a feed-pump discharge check valve was repaired.

**March 1968 (Ref. 12)**

A rod-position-indicating coil stack was replaced.
One of the pressurizer electric relief valves was replaced.
Both pressurizer spray-control valves were repacked, and gaskets were renewed.
A surface blowdown line was installed on one of the steam generators.
Isolation valves were installed on the bottom blowdown lines for one of the steam generators.
All upper-level tap isolation valves on the steam generators were cut out, and new welded bonnet valves were installed.

An internal casting crack was ground out and welded on the lower internal casing of one of the main feed pumps.
One of the drain valves in a reheater steam-supply line was cut out and replaced with a welded bonnet.
A new body was installed on one of the feedwater-regulating valves.
A new test motor was installed on one of the turbine-control valves.
A crack in the body of a discharge check valve for one of the heater drain pumps was ground out and welded.
A new inboard bearing was installed on one of the component-cooling pumps.
The steam isolation valve on one of the moisture-separator reheaters was lapped.
The shaft sleeve on one of the heater drain pumps was replaced.
One of the boric acid pumps was disassembled, and a new canned-type motor was installed.
A new 8-in. gate valve was installed in the steam-supply line to two moisture-separator reheaters.

**April 1968 (Ref. 13)**

A stop valve in one of the coolant loops was repacked, and the stem was cleaned.
One of the pressurizer solenoid relief valves was relapped.
Three motor-operated coolant-loop drain valves were lapped.
In one condenser, 127 tubes were plugged, and 14 tubes were rerolled.
All four isolation trip valves in the main steam line were repacked.
The stem for one of the feedwater-regulating bypass valves was cleaned, and the seat was relapped.
One of the control-rod position-indicating coil stacks was replaced.

**May 1968 (Ref. 14)**

A new seal was installed in the auxiliary boiler feed pump.
One of the two control air compressors was overhauled.
Eroded elbows were replaced in the recirculation line for the charging pump.
A leaking low-pressure relief valve on the letdown line was repaired.
A new mechanical seal was installed on the purification pump.
The disks on one of the steam-generator feed-pump recirculation valves were lapped.
A new hydrogen regulator was installed at the turbine-building manifold station.

**June 1968 (Ref. 15)**

The drain-trap bypass valves were replaced in two steam-extraction lines.
A 1-in. equalizing valve between two of the moisture-separator reheaters was replaced.
A feedwater-line-flow $\Delta P$ upstream isolation valve was replaced.
A 4-in. manual isolation valve was installed in the equalizing line for each turbine-stop–turbine-trip valve.
Pinhole leaks in the body of one of the feedwater check valves were repaired.
One of the charging pumps was overhauled, and mechanical seals were installed.
The high-pressure relief valve in the letdown line was replaced.

**July 1968 (Ref. 16)**

Two pumps were installed for seal-water injection into the steam-generator feed pumps.
Both steam-generator feed-pump recirculation-flow control valves were relapped.
Clearances between the disk and strainer support ribs on one of the turbine-control valves were increased, and the ring contact area was ground down.
Two new increased-capacity high-level dump valves were installed on both reheater drain tanks.
A new cam was installed on one of the turbine-control valves to expand the oil-pressure operating band.
A new gasket was installed in the feed-line check valve to one of the steam generators.
A new generator-seal-oil vapor extractor was installed.
A new high-pressure relief valve was installed in the letdown system
A new welded-bonnet valve was installed in the drain line from one of the turbine-stop–turbine-trip valves.

**August 1968 (Ref. 17)**

Seal-adapter rings were installed in one of the feedwater check valves and were seal welded.
An eroded elbow in one of the charging pumps was replaced with a modified piping configuration.
The seal-injection pump for the steam-generator feed pump was repacked, and leak-off piping was installed.
The bonnet gasket on the isolation valve for one of the reheater-drain-tank normal-level control valves was replaced.

**September 1968 (Ref. 18)**

Mechanical seals were replaced on one of the charging pumps.
The bonnet stud nuts were replaced on both pressurizer spray valves, and valve flanges, studs, and nuts were completely welded because of bonnet-gasket leakage.
The reverse rotation pawls on all four reactor-coolant pumps were replaced.
Twenty-four main-steam-system drain valves and instrument root valves of the four-bolt bonnet type were replaced with the welded-bonnet type.
New servomotors, rings, springs, and linkage were installed on two of the turbine-control valves.
The shafts, seal rings, and bushings were replaced in both turbine-stop valves.
A new test solenoid-type dump valve was installed in the autostop oil system for the turbine-stop valve.
The atmospheric-steam dump valve was overhauled and lapped.
A new pressurizer heater controller was installed.

**October 1968 (Ref. 19)**

Temperature-control valves were installed in the cooling systems for both control air compressors.
One of the charging pumps was overhauled.

**November 1968 (Ref. 20)**

The rod-drive motor for one of the control rods was shut down twice and realigned because of vibrations; during the second shutdown the motor was doweled to the base after alignment.

Heat sinks were installed on the fuses in the pressurizer–heater control group in an attempt to stop continuous fuse failures; when the fuse failures continued, the heat sinks were removed, and the heater load was reduced by 25%.
Twenty-two more tubes were plugged in one of the condensers, bringing the total to 158 (see list for April 1968 above).
Leaking and eroded tubes in the moisture-separator reheater were repaired.
One of the condensate pumps was repacked and a new shaft ordered to replace the badly scored one.
The servomotor for the turbine-governor oil system was disassembled, cleaned, and deburred.
Three of the rod-position-indicating coil stacks were replaced.
A leaking bonnet flange on the charging-line check valve in one of the coolant loops was repaired.
The four main-steam-line isolation valves were repacked.
All four isolation and check valves in the auxiliary feed-pump discharge to the bypass lines for the main feed system were replaced.
All four control valves in the main-feed-system bypass line were replaced with uprated, single-seat valves.
High-pressure flexible supply lines were installed to the feed-pump mechanical seals.
Friction-type vibration dampers were installed on two of the feed lines in the containment.

**December 1968 (Ref. 21)**

The failed drive motor for the main-generator disconnect switch was replaced.
The failed sola transformer in one of the vital-bus inverters was replaced.
A new bellows and new internal seals were installed in the 500-psi relief valve in the residual-heat-removal system
Three rod-position-indicating coil stacks were repaired and modified by removing the exterior aluminum cans so the coils could cool more rapidly.
A modified-type fuse was installed in the power supply to the control group for the pressurizer.
The following modifications and repairs were made to the aerated drains evaporator: the internal surfaces of the batch and distillate tanks were coated with epoxy, a strainer was installed in the concentrator level-control float column, and a strainer was installed in the concentrator supply and recirculation line.

**January 1969 (Ref. 22)**

One of the service-water pumps was overhauled, and new bearings were installed.

**February 1969 (Ref. 23)**

Leaking intake and exhaust valves on both control air compressors were replaced.
An in-line boron analyzer was installed in parallel with the letdown-line radiation monitor to provide continuous read-out of the boron concentration in the coolant.
Miscellaneous oil leaks in the motor-bearings oil-supply and -return lines on both feedwater pumps were plugged.
Automatic casing vent valves were installed on the service-water pump.
Antitwist devices were constructed and installed in the 10 main steam-dump valves to prevent their twisting during operation.

**April 1969 (Ref. 25)**

Two of the turbine-control valves were modified as follows: The strainer for each valve was replaced with a combined muffler–strainer with the lower few inches thickened and drilled with holes to accommodate a pressure drop across the valve at reduced valve openings; also, chamfered rings, heavier operating linkage, heavier springs, and a large servomotor were installed in each valve.
For each turbine-stop valve, the seat part was enlarged, and a new seat with less clearance was installed.
The gland faces of the high-pressure turbine-seal steam glands were remachined, and drain parts were added to prevent condensate cooling.
Eighty tubes in a second moisture-separator reheater were repaired, and 11 inaccessible tubes were plugged.
The thrust bearing in one of the condensate pumps was replaced, and the shaft was repaired.
A crack in the outer casing of one of the heater drain pumps was repaired.
The mechanical seals on both feedwater pumps were replaced.
A second-stage impeller on one of the feedwater pumps had shifted and was grinding against the pump casing; the impeller was removed, and a new one was installed and then secured in place by a spacer.
Seventeen more tubes were plugged in the leaky main condenser, bringing the plugged tube total to 175 (see lists for April 1968 and November 1968 above)
A motor operator was installed on the auxiliary feed-pump discharge to the containment.
One of the charging pumps was overhauled.
The bellows in the high-pressure-relief valve in the letdown line was replaced.

**May 1969 (Ref. 26)**

The seals on one of the primary water transfer pumps were replaced.
A new stem extension was installed in the drain-header motor-operated valve.

**June 1969 (Ref. 28)**

New mechanical seals were installed in pumps for the aerated-drains evaporator-distillate system and for the batch tanks.
Plant-fabricated stems and plugs were installed in all four feedwater-flow-control valves.
Three rod-position-indicating coil stacks were replaced.
The 6-in. steam-supply pressure-equalizing valve between two of the reheaters was replaced.
A new bonnet gasket was installed in the discharge isolation valve for the heater drain pump.
New mechanical seals were installed in one of the steam-generator feedwater pumps.
The exhaust pipe for the compressor for the control-air system was enlarged, and the exhaust valves were overhauled.
Leaks in the bearing-housing seal at the turbine end of the generator were repaired.

**July 1969 (Ref. 29)**

New mechanical seals were installed in one of the reactor-coolant pumps and in one of the feedwater pumps.

The upstream 6-in. isolation valve in the drain line for the reheater drain tank was replaced because of a pinhole body leak and a leaking bonnet gasket.

**September 1969 (Ref. 31)**

Thimble plugs were installed in five leaky feedwater-heater tubes.
The rod-control slave-cycler clutch was replaced in one of the control-rod subgroups.

**October 1969 (Ref. 32)**

Twenty-seven tubes in a feedwater heater were plugged.
The mechanical seals were replaced in one of the feedwater pumps, and the pump was dynamically balanced.
A leaking flange on a discharge check valve on one of the feedwater pumps was welded.
A new bonnet gasket was installed on the discharge valve on a heater drain pump.
A motor-operated isolation valve was installed in the common steam supply to two reheaters.
A slave-cycler clutch on one of the control-rod subgroups was replaced.

**November 1969 (Ref. 33)**

The turbine overspeed-trip set point was reduced from 5% to 2% to decrease the maximum overspeed.
A new motor was installed on one rod-drive cooling fan, and all other motors were lubricated.
New oil-dump valves of the solenoid type were installed in the turbine governor.
The impeller sleeve on the turbine governor was replaced.

**December 1969 (Ref. 34)**

The pump bearings in the main stack radiation monitor seized after overheating; the compressor was rebuilt, and new belts were installed.
The leak-off piping for one of the heater-drain-pump glands cracked at the threaded section to the gland and was replaced with a new pipe nipple.

**January 1970 (Ref. 35)**

A new steam-generator feed pump was installed and tested.
Three leaking tubes in one of the control-air-compressor aftercoolers were plugged.

**February 1970 (Ref. 36)**

New diaphragms were installed in two ion-exchange drain header valves.
Temporary transformers were installed in the transformer for the data logger power supply; the original transformer had failed because it was overloaded.
A leaking bottom flange in the normal-level control valve of the reheater drain tank was repaired by welding and machining the flange.

**March 1970 (Ref. 37)**

A new control valve was installed in the generator hydrogen cooler.
Steam cuts on the discharge flange and the upper discharge column of the heater-drain pumps were repaired.
A new valve body and plug were installed in one of the feedwater-control valves.
Leaking gaskets were replaced in one of the reheater normal-level valves and one of the feedwater-control valves.
A thrust bearing in one of the heater-drain-pump motors was replaced.

**April 1970 (Ref. 38)**

A new permanent transformer was installed in the power supply for the data logger.
New, larger impellers were installed in both condensate pumps.
A new packing gland for one of the main-steam-line nonreturn valve stems was installed, and the scored valve shaft was polished.
The tube bundles for one of the moisture-separator reheaters were replaced.

**May 1970 (Ref. 39)**

All four overpressure-relief isolation valves in the coolant loops were replaced with integral back-seat valves.
All four circulating-water pumps were repaired.
Crud traps were installed in the upper-level sensing lines for three of the four steam generators.
The 1-in. bypass lines for the main-steam-line nonreturn valves were replaced by 2-in. bypass lines.
New rod-position-indicating coil stacks were installed.
The control rods were repaired as follows: all slave-cycler clutches were replaced, one of the slave-cycler motors was replaced, the rod-group selector was replaced, and new bypass switches were installed around the time-delay relays for the half-power contactors.
New surge capacitors were installed on both charging pumps.
All main-steam-line dump valves were relapped.
A cracked bus bar on one of the 4160-V electrical buses was replaced.
New wear rings were installed on the low-pressure section of the safety injection pump.
The low-side bushing gaskets on the main transformer were replaced because of excess nitrogen leakage.
The 4-kV-bus grounding equipment was installed.
A new seat was installed in one of the feedwater-control valves.
A new low-pressure safety injection pump was installed.

**June 1970 (Ref. 40)**

Three leaking tubes were plugged in one of the fan-type coolers for the containment recirculating system.
A worn thrust bearing was replaced in one of the service-water pumps.
Eight feedwater-heater tubes were plugged.
Cracked baffle plates in the main condenser were repaired.
Three new all-stainless-steel pressurizer safety valves were installed.
The pressurizer spray valves were repacked.

**July 1970 (Ref. 41)**

The rotor, stator, and bearings were replaced in the boric acid pump.

A new feedwater-control valve was installed on one of the auxiliary heating boilers.

**August 1970 (Ref. 42)**

A new inboard bearing was installed on one of the steam-generator feed pumps.
One of the control air compressors was overhauled.
The bushings of one of the turbine-stop valves were honed, and a new valve stem was installed.
Gland leak-off piping for the low-pressure safety injection pumps was installed.

**September 1970 (Ref. 43)**

Three rings of packing were added to all loop bypass valves.
The hot-leg isolation valves for three of the four coolant loops were repacked.
One of the heater-drain pumps was replaced.
Leaking gaskets were replaced on one of the pressurizer safety valves and on the leak-off header-valve stem of the isolation-trip valve
A new bonnet gasket was installed on one of the steam-generator blowdown valves.
A leak on one of the feedwater-flow transmitter lines was repaired.
A small leak on one of the feedwater check valves inside the containment was repaired.

## Instrumentation and Control

The information listed below does not include calibration and testing activities.

**July 1967 (Ref. 4)**

All four power supplies for the power-range channels were replaced.
A defective alarm unit in the reactor-control and -protection circuit was replaced.
Four rebuilt Thermovolt relays were installed on the lower-bearing-temperature channels for the reactor-coolant pump.
A faulty $\Delta P$ transmitter in one of the pressurizer-level channels was replaced.

**August 1967 (Ref. 5)**

Bias power supplies were installed in the low-pressure trip-calculator units.
New resistance converters were installed in the loop $\Delta T$ inputs to give a better signal-to-noise ratio.
A new condensate pot was installed on the reference leg of the level transmitter in the level control of the heater-drain tank.

**September 1967 (Ref. 6)**

Pressure gauges were installed in the reheater steam-supply piping.
Air-closure solenoids were installed on all four main-steam-line trip valves.
New power supplies were installed in the steam-generator feedwater controls.

**October 1967 (Ref. 7)**

One of the source-range channels was repaired.
New resistors and wiring for the low-pressure trip calculators were installed.
The high-$\Delta P$ flow transmitters in the main steam line were repaired and recalibrated.
The narrow-range level transmitter in one of the steam generators was repaired.
The amplifier for the pressurizer-pressure transmitter was replaced.

**November 1967 (Ref. 8)**

Four low-pressure turbine-crossover gauges were replaced with compound pressure gauges.
Relays for the vital-bus backup power supply to the feedwater-control system were replaced.
Two new 4-in. reheater-drain-tank level-control valves were installed.
The controls for the water-treatment caustic-proportioning pump were repaired.
An automatic-rod-control unit was replaced with a Sigma MA relay circuit.

**December 1967 (Ref. 9)**

The damaged level-float chambers for one of the reheater-drain tanks was replaced.
A defective transformer in an $R/E$ converter for letdown-temperature input to the data logger was replaced.
An integrator in one of the coolant-loop feedwater flowmeters was repaired.
A faulty amplifier in one of the steam-line differential-pressure transmitters was replaced.

**January 1968 (Ref. 10)**

Stem lubricators and stem-leak-off isolation valves were installed on all steam-dump valves.
New thermocouples were installed in feedwater-heater drains.
The level indicators in both boron-waste-storage tanks were replaced.
A valve positioner was installed on the new temperature-control valve in the component-cooling system.

**February 1968 (Ref. 11)**

A wide-range $R/I$ converter and a $T_{av.}$ $R/I$ converter were replaced in two separate coolant loops.
The 24-V power supplies in the pressurizer-pressure and -level controllers were replaced.

**March 1968 (Ref. 12)**

A new amplifier was installed in the pressure transmitter for one of the pressurizers.
All 28-V d-c power supplies were replaced on all nuclear instrumentation channels.
A booster unit was installed between the valve positioner and diaphragm of the level-control valve for the heater-drain tank, and a solenoid valve was installed between the booster relay and the valve diaphragm.

A main-condenser low-vacuum mercoid switch was installed to operate the steam-dump permissive circuits and the main control-board annunciator.
A new resistance thermometer was installed in the lower bearing of one of the coolant pumps.

**April 1968 (Ref. 13)**

New resistance thermometers were installed on all four main circulating-water outlet lines.
A failed relay in the power supply for one of the feedwater-control circuits was replaced.
A failed 65-V d-c multiple power-supply unit for the data logger was replaced.
A failed power supply in the charging-flow controller was replaced.
A new solenoid was installed on the heater-drain-pump discharge-control valve.
A failed $T_{av.}$ $R/I$ converter was replaced in two of the coolant loops.

**May 1968 (Ref. 14)**

Two rod-position-indicating coils were repaired.
A main-condenser vacuum gauge was installed at the air-ejector station.
All four main-steam-line $\Delta P$ flow transmitters were replaced.
A failed water-treatment-plant flow integrator and a level transmitter in the boric acid mix tank were repaired.
A damaged torque-tube assembly on one of the reheater-drain-tank level controllers was replaced.
The 24-V d-c power supplies to the pressurizer pressure- and level-recorder controllers were repaired.
One of the power-range channel drawers was replaced.

**June 1968 (Ref. 15)**

An improved digital voltmeter for rod-position indication was installed in the main control board.
The temporary 24-V d-c power-supply unit in the charging-flow controller was replaced.
A low-volume pilot was installed in the discharge-valve positioner for the heater-drain pump.
A failed thermocouple on the journal bearing for one of the steam-generator feed pumps was replaced.
The radiation-monitoring detector for the air-ejector discharge was replaced.

**July 1968 (Ref. 16)**

A new mercoid switch for temperature alarms was installed on the discharge side of one of the control air compressors.
A capacitor, which had failed across a 24-V d-c relay, was installed in the power-supply unit for the pressurizer-pressure and -level controllers.
A back-pressure gauge was installed on one of the condensers.

**August 1968 (Ref. 17)**

New hydropneumatic recirculation-flow-control transmitters for the steam-generator feed pump were installed.
Two vacuum manometers were installed on the main condensers.
A chattering rod-bottom-bistable relay was replaced.
Local 0- to 3000-psi discharge-pressure gauges were installed on both charging pumps.
A cracked air manifold on the high-level dump-valve positioner for one of the reheater-drain tanks was replaced.
New differential-pressure control valves were installed in the new seal-injection system for the steam-generator feed pump.
A failed temperature detector for the lower bearing in one of the reactor-coolant pumps was replaced.
The resistance thermometer in the spray line for one of the coolant loops was replaced.

**September 1968 (Ref. 18)**

The RTD in the $T_{av.}$ circuit for one of the coolant loops was replaced.
The $R/I$ converter for the $T_{av.}$ circuit in one of the coolant loops was replaced.
Another Zimmerli gauge was installed on one of the condensers.

**October 1968 (Ref. 19)**

Failed $R/I$ converters were replaced in the $\Delta T$ circuitry for two of the coolant loops.

**November 1968 (Ref. 20)**

The failed wide-range level indicator on one of the steam generators was replaced.
A failed slide-wire resistor in the $T_{av.}$ indicator for the main control board of one of the coolant loops was replaced.
Three coil stacks for control-rod-position indication were replaced.
One of the power-range channel drawers was replaced because of a dead data-logger signal and a poor connection to the overpower recorder.

**December 1968 (Ref. 21)**

Three power-range channel drawers were replaced.
The capacitors in 13 electronic alarm units within the plant-protection and -alarm circuitry were replaced with upgraded ones; these units are located in the steam-dump actuation circuit, the automatic-rod-control circuit, the rod-stop permissive circuit, the low-pressure trip-calculator output circuit, and several protective alarm circuits.
The $T_{av.}$ and $\Delta T$ circuits for the coolant loop were modified by the replacement of five high-failure-rate capacitors with higher quality capacitors.
Three seal-in-type relays were installed in the shunt trip circuit of one of the trip breakers.
On two different power-range channel drawers, capacitors were added between the drawer test switch and ground to prevent voltage transients during testing of the rod-drop–rod-stop circuit.

**January 1969 (Ref. 22)**

Twelve more capacitors in the electronic alarm circuits were replaced with upgraded ones.
Four more $T_{av.}$ and $\Delta T$ circuits for the coolant loop were modified by the installation of five capacitors of higher quality in each unit.
A 120-in.-Hg manometer was reinstalled in the containment-leak-monitoring system.

Pressure gauges were installed in the tube bundles of one of the reheaters for $\Delta P$ evaluation.
A failed thermocouple was replaced in the outlet line for one of the moisture-separator reheaters.
Microswitches were installed to monitor the open position of the turbine-stop valves.
A proximity switch that closes when the turbine-trip solenoid is energized was installed.
Eight capacitors were replaced in the secondary filtering and regulating circuitry for three of the vital-bus Sola transformers.
A transducer was installed to initiate a data-logger message when the turbine autostop oil pressure is reduced to 45 psig.
One of the electronic feedwater-flow transmitters was replaced.

**February 1969 (Ref. 23)**

A 20-conductor cable was installed from the main control board to the data logger to establish printouts of ion-chamber currents in the power-range detector.
A faulty oil-pressure switch on one of the control air compressors was repaired.
Two cracked fittings on the hydrogen regulator for the generator were replaced.
Two overpower relays for the rod-stop meter were replaced.

**March 1969 (Ref. 24)**

A cracked yoke on one of the feedwater-heater high-level dump valves was replaced.
The $R/I$ converter for the $T_{av.}$ circuit in one of the coolant loops was replaced.
A 0- to 0.5 gal/min flowmeter was installed in series with the boron analyzer.

**April 1969 (Ref. 25)**

The take-up reel for one of the in-core flux-mapping detectors was repaired.
A three-mercoid-logic tripping scheme for plant trips originating from the turbine autostop oil-pressure system was installed.
A high-pressure cutoff was installed for the backup-heater groups of the pressurizer.
$^{16}N$ detectors were installed on the hot legs of two of the coolant loops, and an accelerometer was installed on a core-deluge pipe at the vessel head.

**May 1969 (Ref. 26)**

The narrow-range level transmitter was replaced on one of the steam generators.
A resistance thermometer was replaced in the $\Delta T$ cold-leg circuit of one of the coolant loops.
The feedwater-regulating valves were repacked.
Isolation transformers were installed between the output of the power-range detector and the data logger.
The positioner on one of the feed-control valves was reinstalled.

**June 1969 (Ref. 28)**

A failed transformer in one of the ($\Delta T$-current)/(current-repeater) circuits was replaced.
A broken flexible S tube in the pneumatic feedwater-flow detector for one of the coolant loops was replaced twice during the month.
A pressure detector, a transmitter, and an indicator for the common discharge from the auxiliary pumps were installed.

**July 1969 (Ref. 29)**

Another failed S tube was replaced in a coolant-loop feedwater-flow detector.

**August 1969 (Ref. 30)**

Three resistor time-delay relays for two of the control-rod subgroups were replaced.
An RTD in the $\Delta T$ circuit was replaced for two different coolant loops.
One of the source-range channel drawers was replaced.

**September 1969 (Ref. 31)**

The RTD in one of the coolant-loop cold legs was replaced.
A pressure transmitter and recorder were installed to monitor feedwater pressure in one of the coolant loops.
One of the nuclear channel drawers was replaced because of problems with the load-runback circuitry.

**October 1969 (Ref. 32)**

A flux wire and its take-up reel were installed for the second in-core flux-thimble path.
The feedwater control valves were repacked, and the snubbers were filled and adjusted.
An antirotation device was installed on the discharge valve to the heater-drain pumps.
A 1-sec time-delay relay was installed on the main-generator lead-bus duct.

**November 1969 (Ref. 33)**

A fractured S tube on the feedwater-flow transmitter was replaced.
A faulty RTD bulb was replaced in one of the coolant-loop hot-leg $T_{av.}$ circuits.
Two of the feedwater-flow integrators were repaired.
A transformer and a capacitor were replaced in one of the coolant-loop $T_{av.}$ isolation repeaters.

**December 1969 (Ref. 34)**

A new low-range waste-liquid-effluent flow-control orifice was installed.

**January 1970 (Ref. 35)**

A faulty circuit board in the seal-water flow-indication circuitry for one of the reactor-coolant pumps was replaced.

**March 1970 (Ref. 37)**

A new outlet valve was installed in the hydrogen cooler for the generator.
An amplifier in the pressurizer-level indicating circuit was replaced because of erratic drifting.

**April 1970 (Ref. 38)**

A new plug was installed in one of the feedwater-control valves.
A new 24-in. operator was installed in the heater-drain discharge valve.
Connectors for the rod-position-indicating coil stack were replaced.

**May 1970 (Ref. 39)**

Five failed in-core flux thimbles were removed from the core.
All cables for the detectors in the nuclear instrumentation system were changed.

**June 1970 (Ref. 40)**

A new low-pressure flow indicator was installed in the safety injection pump.

**July 1970 (Ref. 41)**

Two new flux-mapping fission counters were installed.
A steam-generator level transmitter that leaked was repaired.
New oil filters were installed on the air-supply system to the diesel generators.
The thermocouple on the inboard-motor bearing in a steam-generator feed pump was replaced.
A faulty relay in one of the level transmitters in the heater-drain tank was replaced.

**August 1970 (Ref. 42)**

The thermocouple on the inboard-motor bearing in a steam-generator feed pump was replaced.
Two leaking steam-dump valves were lapped.
The totalizing flow integrator in the boron-recovery system was repaired.

**September 1970 (Ref. 43)**

Two RTDs that monitor primary-system temperature were replaced.
The force-balance motor in one of the pressurizer-level transmitters was replaced.
The valve-positioner pilot valve for the cooling water of the nonregenerative heat exchanger was replaced.
A faulty capacitor in the $T_{av.}/T_{ref.}$ alarm unit was replaced.
The flux-mapping-system limit switches, which allow withdrawal in the automatic mode, were replaced.

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# Summary of Conference on Power-Reactor Systems and Components

By Myrna L. Steele*

The Conference on Power-Reactor Systems and Components, held in Williamsburg, Va., Sept. 1–3, 1970, was the first topical conference to be held by the Power Division of the American Nuclear Society (ANS). Abstracts of the papers were published in the Supplement to Volume 13 of the ANS *Transactions* and are available from ANS, 244 Ogden Avenue, Hinsdale, Ill. 60521.

The summaries below are based on a combination of this Supplement, some of the papers themselves, and my own notes taken at the meeting. Readers who are interested in more detail should consult the authors of the papers. Also, a review of the safety-oriented papers that were presented at the meeting is included in *Nuclear Safety*, 12(2), March–April 1971.

## QUALITY ASSURANCE AND IN-SERVICE INSPECTION

Selig, Hedden, and Kosky[1] described an in-service inspection program for a pressurized-water reactor (PWR) based on the guidelines of Section XI of the *ASME Boiler and Pressure Vessel Code*. With the basic provisions of the Code as a guide, they constructed an inspection program aimed at minimizing personnel and reactor-plant downtime. Considered were the requirements for extensive automated equipment for remote manipulation in relation to the overall plant cost, the estimated manpower requirements, and the personnel exposures. Those requirements were then compared with a utility's estimate for nonautomated inspection times and costs. The authors suggested that the utility initiate the in-service program when the plant is first ordered so that the nuclear steam-supply system (NSSS) vendor can work closely with the architect–engineer and the utility to devise a program tailored to the utility's specific requirements. Foster[2] discussed the role of the architect–engineer in complying with Section XI of the *ASME Boiler and Pressure Vessel Code* and outlined a set of in-service inspection guidelines to be developed and implemented from the initial review of the reactor manufacturer's proposal through plant lifetime. He listed and discussed the following proposed guidelines for the architect–engineer: evaluation of the reactor manufacturer's proposal, overall plant design, planning the in-service inspection program, and improvement of the in-service inspection program.

Gross and Johnson[3] described an automated ultrasonic inspection system that was designed to operate in the cylindrical core-belt region of the reactor vessel. The inspection system is composed of two ultrasonic transducers that are transported over the reactor-vessel surface in an overlapping square-wave pattern. One transducer acts as a transmitter and the other as a receiver. The amplitude of the received signal is recorded for each set of test-location coordinates for subsequent analysis. The authors reported some signal-dispersion problems for very wavy stainless-steel surfaces.

Bernsen[4] discussed in detail the American National Standards Institute (ANSI) standard for nuclear power-plant quality assurance and outlined the objectives of the standard and described the interconnections between the standard and the AEC General

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Design Criteria. He identified, defined, and discussed 12 principal areas where difficulties are encountered in implementing the AEC General Design Criteria and described the efforts of the standards committee in trying to clarify these problems.

Grob, Corcoran, and Wasilenko[5] described the development of the quality-assurance program at Consolidated Edison's (Con Ed) Indian Point Units 2 and 3 turn-key projects. They discussed the increased emphasis on quality assurance and contrasted the activities for Units 2 and 3. The inspection of potential vendors by Con Ed personnel was described, and the conclusions were discussed. Norris and Jokela[6] discussed the quality-assurance program that Battelle–Northwest developed for Northern States Power Company (NSP). The quality-assurance program was designed to satisfy 10 CFR 50 criteria and a quality-assurance committee that was composed of NSP personnel and NSP-designated architect–engineer representatives. Norris and Jokela defined the types and levels of design reviews and design controls; described the procedures for making design changes; and discussed the detailed instructions that were developed for production methods and processes, the quality-assurance specifications and contract clauses that were evolved for Class I systems and components, and the difficulties and reactions encountered with the suppliers. The authors described the problem of educating all the people who were concerned with a given project and reported that the cost of the quality-assurance program was less than 1% of the total plant cost.

## SAFETY-RELATED SYSTEMS AND COMPONENTS

Carbiener[7] used emergency core-cooling system (ECCS) design to illustrate the applications of analytical models and mockups to reactor-safety research and development. He described the conditions under which safety research must be conducted, the safety limits that can be tolerated by the ECCS during operation, and the analytical models used in evaluating ECCS capability and integrity.

Van Sickel[8] discussed the design criteria, constraints, and goals for equipment supports for an 800-MW(e) PWR station. He described the use of a bottom-support system and the static-load analyses of this system for 15 different cases. Van Sickel considered the overall system costs and the factors that contributed most heavily to the costs. Dainora[9] analyzed the probable dynamic effects of pipe rupture on component-support systems in an 800-MW(e) plant. He described the analytical–computational model used in the analysis of a guillotine break and a longitudinal split and discussed the results. Kadakia[10] described the application of a numerical integration method to the determination of the dynamic response of reactor internal structures under blowdown conditions. He discussed the dynamic model that was formulated, defined the conditions and constraints that bounded the analysis, and outlined the conclusions reached.

Fox, Salvatori, and Thailer[11] discussed the static and dynamic bending tests that were performed on pressurized-water-reactor (PWR) carbon-steel and stainless-steel pipes. They listed and analyzed the preliminary results of these tests. Carlson and Russell[12] discussed the methods used to evaluate the piping integrity at the Hanford K reactors. They described the applied proof-testing conditions, achieved through system overpressure, and covered the data resulting from fracture and stress analyses. The authors then outlined the conclusions of this proof-testing study.

Deackoff, Feinstein, and Vanasse[13] described a computational method for determining the amount of hydrogen that is generated under design-basis-accident (DBA) conditions. They discussed the features of the CORHYD computer program, which computes fission-product inventory as a function of time after shutdown; volume of hydrogen that is generated as a function of time; and volume percent of hydrogen in the containment vessel with one, two, and three recombiners operating.

## CODES, STANDARDS, AND CRITERIA

Rosen[14] discussed the standards program of AEC's Division of Reactor Development and Technology (RDT) for reactor-development projects. He described the RDT Standards Program as a focal point for defining technical development and quality-assurance practices and for establishing uniform safety criteria and practices. Nicholson's[15] topic was the RDT Standards Program as related to the preparation and use of standards for liquid-metal-cooled fast breeder reactors (LMFBRs). He described the procedures by which standards are applied to a reactor component from procurement through fabrication and delivery. Nicholson then discussed the problems associated with the application of a standard and the processes by which the ideal standard is evolved.

Russ[16] discussed the standards activities of the Institute of Electrical and Electronics Engineers (IEEE)

relative to nuclear-power-plant systems and components. He outlined the unique requirements of nuclear systems and noted that, although engineering standards have traditionally evolved in a leisurely manner, nuclear power plants require preoperational standards. Russ stated that acceptance consensus is often difficult to obtain in practice even though consensus means a weighing of objections and not 100% agreement. He discussed the attitudes of older standards personnel with regard to both redundancy and urgency of efforts in nuclear standards preparation and then listed the requirements for standards as (1) thorough research, (2) definition of task, (3) requirements for task, and (4) a written charter. Russ suggested that nuclear standards be published in so-called editions and that the standards be categorized according either to basic system criteria and imple mentation of these criteria or to equipment and procedures criteria.

Case[17] discussed AEC's efforts in attempting to establish standards and procedures. He described the 1967 reorganization of the regulatory staff to form the Division of Reactor Standards (DRS) and explained how DRS interfaces with the other AEC divisions and the various technical societies that are responsible for standards development. He discussed the general design criteria, which were published in 1965 and revised in 1967 and 1970, and presented the status and current thinking of the AEC on the quality-assurance criteria for nuclear power plants which were published in 1969 as Part B of 10 CFR 50 and the seismic siting criteria that were proposed in 1969.

Joslin[18] discussed standards activities that have, historically, been carried out by industry and described the efforts of ANSI's Nuclear Standards Board in coordinating the current industry-standards programs. According to Joslin, more utility participation is needed in standards work, and more standards are needed by the utilities. Joslin stated that progress in standards development was not adequate and gave several suggestions for greater progress.

## INSTRUMENTATION AND CONTROL

Upton, Brown, and Spear[19] described a self-powered fast-neutron-flux monitor for in-core use in large fast power reactors. The reaction of the $^{9}Be$ in the detector is $^{9}Be(n,\alpha)^{6}He$; the $^{6}He$ provides a current signal to be detected by means of beta particles. The authors detailed results of both numerical and experimental analyses. Porter, Hoitink, and Jackson[20] discussed in-core instrumentation for LMFBR radiation environments and operating conditions. They described the spurious noise phenomena that resulted from cables and cable connectors when they were operated under elevated-temperature conditions and reported some methods that have been useful in mitigating the deleterious effects of breakdown pulse noise. Billeter and Brown[21] described an in-core microwave temperature sensor for fast reactors that operates at high temperatures. They discussed the principles on which the sensor works and the operational characteristics of an experimental model.

Sides[22] discussed the control characteristics of a 1000-MW(e) molten-salt breeder reactor (MSBR). He described the analog model (shown in Fig. 1) that was used to investigate the basic requirements of the MSBR and analyzed the results of this analog simulation. Preliminary analyses of transient effects resulting from the overcooling of the salts were considered briefly.

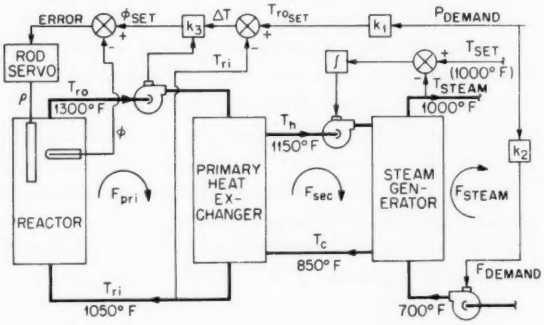


**Fig. 1 The MSBR plant and control-system simulation model.[22] Temperatures are design-point values. $T_{ri}$ = reactor inlet temperature; $T_{ro}$ = reactor outlet temperature; $\phi$ = phase.**

Macrae[23] discussed the control system for the proposed Westinghouse LMFBR demonstration plant. He presented brief descriptions of each of the plant systems, discussed the control strategy that was applied to the systems, and described the instrumentation and data-acquisition systems that feed into the control logic.

Ferguson, Holland, and Ebersole[24] detailed the design criteria for the protection and actuator systems of nuclear power plants. These criteria, which were developed by the ANS Standards Committee, pertain to the integrated response of actuator systems, protection systems, and supporting systems. The criteria provide: design features of the station, principal design

criteria for the protection and actuator systems, safety limits for station operation, definitions of process variables where protective measures should be taken to assure safe station operating conditions, degradation limits of design features for maintaining safety of station operating conditions, inspection criteria, and administrative controls.

## COMMERCIAL REACTORS: COMPONENT DESIGN AND FABRICATION

McDonald and Johnson[25] gave a detailed discussion of a once-through steam generator for a PWR power plant. They described the series of laboratory-scale generators that were used for testing and development of the full-scale steam generator. They detailed the design data, the performance requirements, and the restrictions and constraints imposed by a PWR system and discussed the results of the following: temperature measurements on primary and secondary fluids and on tube metal; simulated emergency transients; and accelerated fouling tests.

Bush[26] discussed the pressure-relief valves, safety valves, and rupture disks that are used in light-water-reactor cooling systems. The requirements for large pressure-relieving capacities in commercial nuclear power plants were considered, and methods for determining the sizes of pressure-relief devices were outlined. Bush described the types of valves for both BWRs and PWRs and discussed the basic features of each. His basic criterion for selection of plant relieving capacity was that the cooling-system pressure must not exceed 110% of design pressure.

O'Mara[27] discussed the use of hydrogen recycle to obviate the need for a catalytic hydrogen recombiner and/or fission-product-decay tanks for storage of waste gases that are stripped from the coolant of a load-following PWR. The design premise underlying hydrogen recycle is that the gaseous fission products are removed only at the steady-state full-power operating condition. O'Mara included examples to illustrate the principle and its applications.

Meyer and Hekkala[28] described a coolant radiation monitoring system (Fig. 2) for large PWRs which provides a check on coolant-activity levels during operation. The shielded scintillation detector continuously measures the gamma activity of water samples drawn from the letdown stream of the cooling system. The authors also described the calibration procedures and discussed the results.

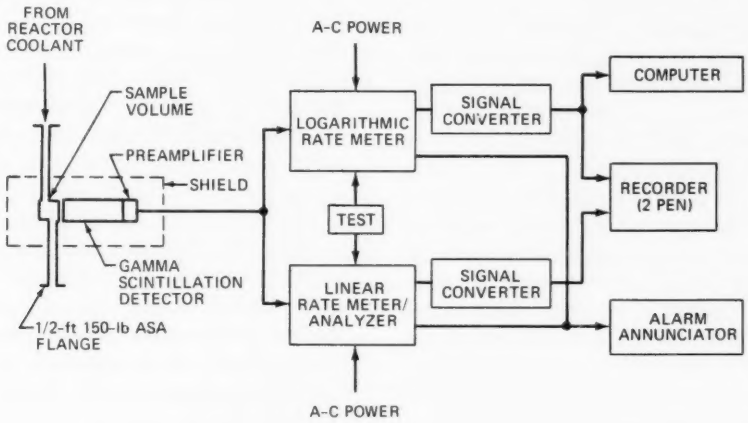


Fig. 2 Block diagram of process radiation monitoring system.[28]

Larsen and Edelstein[29] described the handling system for the primary-coolant filters at Northern States Power Company's twin-unit Prairie Island plant. They discussed the concepts, decisions, and trade-offs that affected the design and briefly considered the problems and constraints that they encountered. Special attention was given to minimizing radiation exposure under high-radiation conditions and to preventing the spread of contamination.

Schmitt[30] described the results of full-scale tests of containment response under simulated DBA conditions

in the Carolinas-Virginia Tube Reactor (CVTR). He briefly described the CVTR containment vessel and the conditions under which the DBA tests were conducted. He then discussed the heat-transfer computer codes and analytical models that were used for the containment-response analysis. The conclusions were: the heat-transfer coefficients were much higher than predicted, the CONTEMPT computer code was verified for DBA heat-transfer calculations, results indicated that development of a multinode heat-transfer code might give even better results, the containment-spray water effectively reduced both the temperature and pressure under DBA conditions, and containment integrity was preserved during and subsequent to blowdown.

Toth and Tunnell[31] discussed recent applications of steel and steel/composite containment vessels for light-water reactors. These applications included containment vessels for the Hutchinson Island, Davis–Besse, and Sequoyah PWR stations and the containment vessels proposed for the Shoreham, Zimmer, and Limerick BWR stations. All containment vessels employ a multibarrier concept that features a primary steel vessel and a secondary concrete-shield building. The authors described the containment vessels for the PWRs in detail and discussed the containment-vessel construction. The unusual vessel shape required for BWRs and the proposed over/under configuration for the reactor plants mentioned above were considered briefly.

## ADVANCED REACTORS: SPECIAL DEVELOPMENT PROBLEMS

Goodjohn and Trauger[32] described the efforts exerted in developing a technology for the high-temperature gas-cooled reactor. Of special interest were the coated-particle-type fuel, the prestressed-concrete reactor vessel, the main helium circulators, the once-through steam generators, the remote handling equipment for irradiated fuel, and the bypass helium-purification system. The authors also briefly described the unique features of each system.

Briggs[33] outlined the operations experience, gained over several years, with the Molten-Salt Reactor Experiment (MSRE) and discussed the more outstanding points. Possible difficulties in extrapolating from the MSRE to a large molten-salt breeder reactor (MSBR) were evaluated. Briggs also discussed the areas of chemistry for the preferred coolant salt (sodium fluoroborate) that needed to be developed, the problems that were encountered in the Hastelloy N in the MSRE, and the improvements that are needed in graphite.

Koch[34] discussed the special development problems related to the core-coolant and secondary systems of the LMFBR. Koch stated that the core must be a mechanical assembly of functioning precision parts and that steam generation with high-temperature sodium involved, primarily, materials and reliability problems. He also pointed out that these problems affect the economics rather than the feasibility of an LMFBR plant.

Guyker and Ver Planck[35] discussed gas-cooled fast-reactor development problems. They drew a parallel between gas-cooled fast-reactor and LMFBR design studies and then specified areas of gas-cooled fast-reactor technology where unique problems exist. The authors stated that the main problems were associated with the high-pressure–high-mass-flow cooling gas circulating through the core. Also discussed were the problems with reactor-physics calculations, the radiation effects on fuel and core materials, and possible problems related to the development of plant components. O'Brien, Burke, and Walker[36] discussed the preliminary results of some loss-of-coolant accident analyses for gas-cooled fast reactors. They described the results of an exponential pressure decay to 15 psia with a 30-sec time constant and showed the percent coolant-mass flow and fuel-cladding temperatures as functions of time.

Pease[37] described the development problems that are anticipated for advanced CANDU systems. He discussed the efforts to understand the mechanism of radioactive mass-transport problems in the cooling systems because the high-radiation areas pose maintenance problems. Pease also considered the various types of coolants that could be used in the CANDU reactor.

## COMMERCIAL REACTORS: CURRENT EXPERIENCE

Hartley[38] detailed the coolant-activity problem that was experienced from April 1969 until refueling in April 1970. He explained how the coolant activity had leveled off at about $1.1 \times 10^{-1}$ $\mu$Ci/ml while the reactor was operating at 490 MW(e), decreased during the first 2 days of the April 1969 maintenance shutdown to about $3 \times 10^{-3}$ $\mu$Ci/ml, and then rose to about $1.1 \times 10^{-1}$ $\mu$Ci/ml at the conclusion of cool-

down. He described the contamination of the primary-system cleanup devices after startup and discussed the later problems of operating with 12 to 15 $\mu$Ci/cm$^3$ $^{133}$Xe levels in the containment vessel. Hartley then outlined the experiences in refueling during the April 1970 shutdown and described the efforts that are under way to identify the source of the $^{58}$Co contaminant.

Montgomery[39] summarized problems that had been encountered with safety and relief valves in PWRs. He considered the leakage problems and environmental conditions under which these valves must operate and discussed some possible solutions to the leakage problems.

Johnson and Brown[40] described a technique, called the parallel-standoff technique, for plugging heat-exchanger tubes in water-cooled reactor systems. The technique uses an explosive charge in a blind, hollow plug which welds the plug to the inside diameter of the tube. They considered the standard methods used to plug the heat-exchanger tubes, explained why these are inadequate, and outlined the advantages of this parallel-standoff method.

Falk and Michels[41] described a BWR off-gas system that utilizes catalytic recombiners and charcoal adsorbers. The recombiners remove the radiolytic hydrogen and oxygen, and the charcoal adsorbers delay passage of noble gases to allow the krypton and xenon to decay and the resultant decay products to be deposited on the adsorbers. They discussed the radioactive-effluent limits as prescribed by 10 CFR 20 and related these to the new off-gas-system design.

Hathaway, Mehann, and Parker[42] told of the experience gained from the first refueling of the N.S. *Savannah* during the fall of 1968. After a brief description of the reactor, the authors detailed the procedures and the organization that were established for refueling the reactor and the refueling process. They explained the methods by which personnel exposure was controlled and discussed the overhauling of the control-rod-drive system and the waste-handling methods. The authors briefly summarized the problems that were encountered during refueling and then briefly outlined the physics tests that were conducted at dockside and during sea-trial testing.

## ADVANCED REACTOR COMPONENTS: MATERIALS AND DESIGN

Feldman[43] discussed the application of remote handling techniques to: postirradiation handling and analysis, remote fuel processing and fuel fabrication, and reactor operation and maintenance. He described the equipment that has been developed and the fundamental design criteria for remote handling facilities. Feldman then discussed the activities of the Fuel-Cycle Facility, which provides remote fuel processing and fabrication for Experimental Breeder Reactor II (EBR-II). He described the exposure and burnup characteristics of the EBR-II fuel and summarized the current design and construction status.

Northrup and Davis[44] described the Fort St. Vrain prestressed-concrete reactor vessel from the design, which started in February 1966, to final concreting, which was completed on Feb. 13, 1970. Problems in establishing design criteria for the prestressed-concrete reactor vessel were discussed, and the requirements were listed. The authors summarized developments in: scale-model testing of the prestressed-concrete reactor vessel, two- and three-dimensional finite-element stress analysis, physical properties of materials for the prestressed-concrete reactor vessel, and environmental and loading effects on instrumentation, prestressing system, tendon-anchor zones, and liner. The authors explained the construction techniques used for the prestressed-concrete reactor vessel and discussed the methods for maintaining vessel temperatures. Yampolsky[45] summarized the development of the helium circulator for the Fort St. Vrain Station. He discussed the development program on the steam-turbine-driven helium circulator from its inception in 1966 to its completion in 1970, described the mockup-test facility that was used to provide data on individual components and on a full-scale prototype, and presented some typical operations data. Difficulties encountered during the development of the helium circulators were also discussed.

Davis and Perry[46] presented highlights of operating experience with sodium pumps since the first operation of the EBR-II primary sodium system in April 1963. The sodium pumps, for which they presented brief operating histories, are: the two 4670 gal/min single-stage centrifugal primary pumps, a 6500 gal/min a-c linear-induction electromagnetic (EM) secondary pump, a 575 gal/min d-c EM primary auxiliary pump, a 96 gal/min d-c EM pump and a 100 gal/min a-c EM Faraday pump that are used in the primary purification system, two 60 gal/min a-c linear-induction EM secondary-system recirculation pumps, the three small direct-current EM pumps that are used in the primary-sampling and plugging-loop systems, and the 100 gal/min d-c EM pump used in the fuel-element-rupture detection system.

**Table 1 Results of Sodium- and Water-Leakage Testing as a Function of Fuel-Nozzle Diametral Clearance**

| Module No. | 2 | 3 | 4 | 5 | 6 |
|---|---|---|---|---|---|
| Nominal diametral clearance, in. | 0.005 | 0.010 | 0.010 | 0.025 | 0.010 (with seat) |
| Material reference | | | | | |
| Outer cylinder | 304 S.S. | 304 S.S. | Inconel 718 | 304 S.S. | 304 S.S. |
| Inner cylinder | 304 S.S. | 304 S.S. | 304 S.S. | 304 S.S. | 304 S.S. |
| Water leakage results (Temp., 210°F) ($\Delta P$, 140 psi) | 1.95 gal/min | 5.8 gal/min | 4.7 gal/min | 23.5 gal/min | 4.0 gal/min |
| Sodium leakage results (Temp., 900°F) ($\Delta P$, 140 psi) | 2.4 gal/min | 5.6 gal/min | 2.8 gal/min | 15.5 gal/min | 4.0 gal/min |
| Sodium leakage results (Temp., 750°F) ($\Delta P$, 140 psi) | 1.8 gal/min | 5.4 gal/min | 3.3 gal/min | 14.5 gal/min | 3.8 gal/min |

Oakley, Webb, and Wett[47] analyzed results of sodium-flow tests on some fuel-assembly mockups for the Fast Test Reactor (FTR). They described the test program for determining the extent of nozzle erosion caused by flowing sodium at velocities approaching 75 ft/sec. Table 1 shows the results of both the sodium- and water-leakage testing as a function of fuel-nozzle diametral clearance. The authors discussed a series of flow-reduction anomalies which occurred during testing.

Bauer and Conner[48] discussed the materials problems encountered in the design of the LMFBR. The constraints imposed by high temperatures, the flowing-sodium environment, and high fast-neutron fluences were considered. The authors summarized the problems that are peculiar to the various systems and components, such as the piping, valve bodies, and steam generators. In addition, the authors discussed the problems associated with sodium corrosion and with mass and carbon transfer and briefly summarized specific uses of the various types of steels in LMFBRs.

Olson[49] described the in-line vacuum-distillation sodium sampler (Fig. 3) that has been developed for the EBR-II. He discussed the design, outlined operating procedures, and summarized the EBR-II operations experience with the sampler.

Balent and Hirst[50] summarized their efforts toward the design of fuel elements for the fast breeder reactor (FBR). They discussed the results of tests conducted to evaluate the discrete-grid-spacer and the helical-wire-wrap fuel-element designs. The authors described the hydraulic and heat-transfer characteristics of these designs and listed the typical design data that resulted from the developmental efforts.

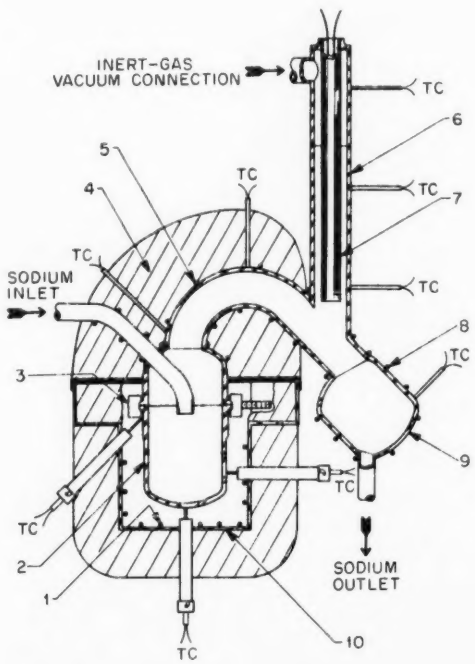


**Fig. 3 Vacuum-distillation sodium sampler.[49]** *TC* = thermocouple.

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These references are to the abstracts of papers presented at the American Nuclear Society's Conference on Power-Reactor Systems and Components, held Sept. 1–3, 1970, in Williamsburg, Va. The abstracts appear, on the pages cited, in the Supplement to Volume 13 of the *Transactions of the American Nuclear Society*.


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12. P. A. Carlson and A. Russell (DUN), Application of Proof Testing To Assess K-Reactor Piping Integrity, p. 10.
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# Sodium-Cooled Reactors: Current Status of Plant-Systems Operations

By Myrna L. Steele*

## FERMI FAST BREEDER REACTOR

Preparations for reactor restart continued[1-7] through early July 1970 with some cold trapping being necessary to maintain primary-system sodium temperatures at <420°F. Criticality was achieved, for the first time since Oct. 5, 1966, on July 18, 1970, at 1104 hr with an excess reactivity of 31 cents at 471°F and with 100 fuel subassemblies in the lattice. There were a total of 22 reactor startups from July 18 through July 31 with no unscheduled scrams.

### Sodium Systems

**Cold Trapping.** About 31 hr of cold trapping[1] during January 1970 kept the primary-system temperature between 398 and 415°F and the maximum saturation temperature at <265°F. The secondary sodium loops were not cold trapped, and the maximum system-saturation temperature was 330°F in loop 2. During February about 50 hr of cold trapping maintained the primary-system temperature between 398 and 420°F and the maximum saturation temperature at <250°F. In loop 2 of the secondary system, with no cold trapping, the maximum plugging temperature was 300°F. With no cold trapping in March, the primary-sodium-system temperatures[3] varied between 400 and 428°F and the maximum saturation temperature was <240°F. Also in March, with no cold trapping in the secondary system, the maximum plugging temperature rose 5°, to 305°F, in loop 2. In April, cold trapping[4] was required for 173 hr to keep the primary-system temperature between 408 and 430°F and the maximum saturation temperature at 370°F. Loops 1 and 3 of the secondary sodium system were cold trapped for 34.4 and 40.3 hr, respectively; loop 2 was not cold trapped. The maximum saturation temperatures for loops 1 and 3 were 285 and 260°F, respectively. During May 1–10, the primary-system temperature[5] was 410 to 430°F; for the rest of May, with 163 hr of cold trapping and a maximum saturation temperature of <240°F, the primary system was kept between 455 and 475°F. No cold trapping was carried out in loops 1 and 2 of the secondary sodium system during May, and only 13 hr of cold trapping was required to keep loop 3 with a maximum temperature of 275°F. The primary-system temperature[6] was 455 to 478°F during June 1–24, 515°F June 25–27 for some preoperational tests, and back to the June 1–24 temperature range for the remainder of the month. The secondary sodium loops were not cold trapped during June, and the maximum saturation temperature was <240°F. Secondary sodium loop 1 was cold trapped for 26.5 hr, loop 2 for 75.5 hr, and loop 3 for 21.75 hr; the maximum saturation temperatures were 300°F for loops 1 and 3 and 350°F for loop 2.

**Primary Plugging Indicator.** A strong indication[4] of trapped gas in the upper part of the primary plugging loop led to the venting of the loop on April 21. In May a gas-flush test[5] was performed to see if an increase in pump power could eliminate the gas. No appreciable changes in flow noise and pumping characteristics were noted.

**Maintenance and Repairs.** Installation of thermocouples[1] in the siphon-breaker lines was finished[2] in February 1970. The thermocouples will be used to monitor sodium flow in the siphon lines. Check-out[3]

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of the 14 thermocouples showed that the response of 3 of them was affected by their physical location, and these 3 had to be moved.

Four defective[1] bearings in the magnetic coupling of the No. 2 overflow pump were replaced in January 1970 to stop the excessive vibration of the pump.

In February the No. 1 primary pump[2] was shut down so that worn shaft seals and thrust bearings could be replaced. In April, electrolyte header leaks had to be repaired in two of the liquid rheostats for the primary pumps, and in June the pump[6] was disassembled for replacement of the argon seals and for investigation of an erratic flow switch in the liquid rheostat. The shaft thrust bearings[3] on the No. 2 primary pump were replaced in March. These bearings had been inspected, greased, and transposed in October 1969. In May 1970 both thrust bearings in the No. 3 primary pump were replaced when a small oil leak in the oil reservoir for the pony motor led to the discovery of a damaged bearing race.

During a siphon-breaker-line flow test[4] on April 1, excessive torque was required to operate the No. 1 throttle valve. Dirt had accumulated in the close clearances between the shaft and plug bore where the valve shaft extended through the floor. The clearances were increased to avoid binding.

### Fuel-Element Handling

During January–June 1970, emphasis remained on the fuel-handling activities[1-7] of the startup preparations. By the end of June, 93 enriched core subassemblies had been loaded into the lattice and 7 central inner-radial-blanket (IRB) subassemblies remained to be replaced by enriched fuel so that criticality could be achieved at the predicted loading of 100 subassemblies.

The first enriched core subassembly[2] that had been fitted with a short flow guard was inserted into the core lattice in February and immediately withdrawn to the transfer rotor because of some count-rate-data anomalies. The removal of this subassembly restored the core loading used for base-count-rate data, so that a new base count could be taken. A new neutron source[3] arrived on March 23 and, after inspection with underwater TV cameras, was inserted into the lattice on March 29.

In April the fuel-element nozzles were tested with the offset handling mechanism (OHM). The nozzle on only one fuel element indicated no spring travel, and that element was moved to another location and the nozzle retested. Unsatisfactory results caused the subassembly to be stored in the transfer rotor for future consideration. In May the subassembly[5] was moved to the transfer tank. By the end of May, all core and IRB positions in the lattice were filled with subassemblies which were not under the hold-down column on Oct. 5, 1966, or which were never operated at significant power levels.

The hold-down mechanism,[5] which had been in the raised position since shortly after the fuel-melting incident in 1966, was aligned and lowered onto its support columns. Strain-gauge measurements ensured that the mechanism would properly engage the support columns in the reactor and that the proper loading force was applied on the subassemblies and columns.

One[6] of the seven IRB subassemblies without flow guards was exchanged with an enriched long fuel subassembly that did have a flow guard, and an approach to criticality was made on June 29. The count-rate data were analyzed by Atomic Power Development Associates, Inc., and the measured reactivity worth agreed with the predicted value. The other six unguarded IRB subassemblies[7] were transferred from the central core region to the Fuel and Repair Building (FARB) transfer tank in July. These subassemblies were replaced with fuel subassemblies, loaded one at a time, and rods were pulled after each loading to check on reactivity.

### Fuel-Transfer Facility

In January 1970 the dashpots of three fuel-transfer pots[3] had to be redesigned and replaced because of problems in guiding the transfer pots into the rotor plate. During replacement operations on January 22, one of the pots was dropped about 20 ft from a transport car onto the transfer-tank rotor when the drip pan for the transfer cask opened. The transfer pot dropped onto an empty source-rod pot. After retrieval of the transfer pot, the fuel-transfer-facility gripper was used as an elevation probe to verify that the plate was not bent. Later the source pot that had received the impact of the dropped transfer pot was visually examined, was found to be acceptable, and was returned to the transfer tank.[2]

On April 9, during fuel-element-transfer activities, the fuel-transfer[4] gripper in the FARB seized because of sodium oxides collecting in the gripper. The gripper was cleaned with alcohol and restored to service. During the transfer of a new core subassembly to the reactor on April 17, one of the front-wheel bearings on the transfer cart froze to the stub shaft, which, in turn, froze to the car frame. The shaft was repaired by machining its outside diameter, and the bearings were

replaced with bearings having a caged-needle design to prevent lateral skewing of the needles. On April 30 the gripper bound again and was again cleaned with alcohol.

In July a failed section[7] of a lead wire on one of the fuel-cask heaters was replaced when the temperature inside the cask could not be raised above 285°F.

### Offset Handling Mechanism

In January 1970, 125 lb of lead was added to the crane-tube counterweight[2] to allow the OHM to be raised to the new elevations required in handling the flow-guarded subassemblies. Subsequent to this weight addition, the machine could not be lowered[3] to its full-down position for nozzle-spring testing. After a comparison of the OHM vertical forces with the spring-plate settings, the current-limit setting of the elevation motor was increased by 10%. Also, an interlock was installed in the azimuth-control circuit to prevent azimuthal movement if the elevation does not correspond with the data-logger program.

### Steam Generators

All three steam generators[1-5] were inverted with nitrogen cover gas through May 1970. In June thermocouples[6] were installed on all three loops to measure sodium inlet and outlet temperatures. A thermocouple was also installed in the feedwater header to measure water-inlet temperatures to the steam generators. A burned-out solenoid coil was replaced on the pressure-source valve for the No. 1 steam generator, and a stuck valve stem was repaired on the feedwater-control valve. These valves had failed to operate during a routine steam-generator isolation test in April, but on June 21, after the repairs, all valves functioned properly in a similar isolation test. In July all three steam generators[7] were pressure tested with nitrogen in preparation for preoperational testing and filling of the feedwater systems. The safety-valve settings and set points were checked for steam generator No. 1. A feedwater main-bypass valve was repacked, and a spare valve plug and spindle were installed in the emergency-cooling steam dump valve.

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# AMERICAN NUCLEAR SOCIETY—CRITICAL REVIEWS

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## ANS GUIDELINES FOR PREPARING CRITICAL REVIEWS FOR *REACTOR TECHNOLOGY*

One interpretation of the words "Critical Review" emphasizes the word *critical.* That type of article would have as its purpose the discussion of a single subject, which has been uncertain and perhaps controversial, in considerable depth. An example might be a critical review of the values of alpha for plutonium, assessing all the work done and arriving at a best current estimate. This paper is primarily addressed to specialists on the particular subject, and serves as an authoritative source for the information they use.

In the second interpretation of a Critical Review, the emphasis is on the word *review,* in the sense of survey. Such a paper would be broader and probably more descriptive in scope. A paper on "Solubility of Metallic Elements in Liquid Sodium" is an example of this category. Such a paper would be addressed to a much broader group of readers and written in a fashion that would be of interest to the majority of reactor technologists. It would provide enough of an introduction that the specialist from another field could immediately appreciate why the subject is important.

The criteria for these two types of articles may not be very different:

1. The paper should be based on a thorough coverage of relevant work on the subject from many sources. A review based on the work of only one individual or group is better suited for publication in one of the regular society journals. The review should be critically selective, reporting only the most valid results and indicating why some prior results have a questionable status. The review should call attention to significant gaps where more work is required in the subject of the article.

2. The paper should be timely, on a subject of active current interest.

3. The scope of articles acceptable as Critical Reviews includes all the subject areas identified for *Reactor Technology* (Economics, Physics, Mechanics, Construction, Fuel Elements, Fuel Cycles, Fluid and Thermal Technology, Fuel Processing, Components, Operating Performance), as well as Materials (including Source and Special Nuclear), Environmental Effects, and Effluent Management. Not desired are (1) reports of original research proposed for first publication and (2) review articles directed toward the specialist in the field of nuclear safety (which is covered by the AEC's bimonthly review *Nuclear Safety*).

4. The paper should be organized so that it is of immediate practical use to the readers. Such organization requires: attention to consistent use of units, presentation of important data in summary curves or tables, and a fully adequate bibliography to the original literature. Details on size, honorarium, style, etc., can be obtained from ANS.

# In-Service Inspection of Nuclear Reactor Coolant Systems

By L. R. Weissert*

On Nov. 24, 1969, the U. S. Atomic Energy Commission (AEC) announced[1,2] a proposal to amend its reactor-licensing regulations to include sections on codes and standards as they relate to systems and components important to safety. With one sweeping move the AEC, in its role as regulator of the peaceful atom, riveted the attention of the utilities and the nuclear power industry on this new requirement to comply with recently developed codes of the American Society of Mechanical Engineers (ASME) and the Institute of Electrical and Electronics Engineers (IEEE). Up to that time, it had been the general practice that pressure vessels, piping, valves, and pumps that are part of the reactor-coolant-pressure boundary should, as a minimum, be designed, fabricated, inspected, and tested in accordance with ASME codes. Similarly, it had been recognized that the electrical portion of protection systems should, as a minimum, meet the criteria developed by the IEEE.

The ASME codes and IEEE criteria generally do not become effective for at least a year after their publication for trial use and comment, and then only for contracts entered into after the effective date. Because of the roughly 6-year lead time from contract to commercial operation of a reactor, new codes or revisions would not be reflected in operating experience for many years. The AEC announcement proposed to leapfrog this traditional pattern and impose codes "at the earliest feasible time." Reactors for which construction permits have been issued but which have not been licensed for operation would be required to comply with codes in effect at the time equipment was ordered. For new reactors, components important to safety, regardless of order date, would be required to comply with the more recent revisions of the codes.

The Nov. 24, 1969, press release included a chart illustrating the requirements as a function of time. The chart shows a consistent requirement for use of the ASME Code for In-Service Inspection of Nuclear Reactor Coolant Systems for plants with construction permits issued on or after Apr. 1, 1970. This code is now Section XI of the *ASME Boiler and Pressure Vessel Code*[3] and is the heart of all in-service inspection programs in the United States. Section XI, which will later be analyzed in detail, was developed under sponsorship of the ASME through the American National Standards Institute (ANSI) N45 Committee as a cooperative effort with the AEC.

The *ASME Boiler and Pressure Vessel Code* was first formulated in 1911 as a result of the strong concern by owners, designers, builders, and insurers of boilers and pressure vessels for public safety. This concern was carried into service life in 1919 with the Inspection Code of the National Board of Boiler and Pressure Vessel Inspectors. Almost all 50 American states and all Canadian provinces have passed statutes based on the model Boiler and Pressure Vessel law, which adopts the two codes as the legal basis for regulating the design, construction, and in-service inspection of boilers and pressure vessels.

The development and definition of in-service inspection for nuclear reactors started in 1967 on a number of fronts. Inspection of the Elk River, Indian Point 1, Dresden 1, and N.S. *Savannah* reactors pro-

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vided valuable field experience. The ASME/ANSI N45 Committee began work on the code. The Power Test Code committees started work on codes for in-service testing of pumps and valves, resulting in Power Test Codes 34 and 35. The Edison Electric Institute and the Tennessee Valley Authority set up a joint Steering Committee to direct and conduct research to develop new methods and equipment of inspection of pressure boundaries.

With the subsequent incorporation of these new codes as proposed rules in the *Code of Federal Regulations* since the AEC announcement and the release of Section XI of the *ASME Boiler and Pressure Vessel Code*, the development and application of in-service inspection have accelerated. This article reviews all significant literature to September 1970, presents the major developments and experiences on operating reactors, and explores the implications on reactor system design and plant operation and maintenance.

## ASME SECTION XI

Section XI of the *ASME Boiler and Pressure Vessel Code* applies to the examination and inspection requirements for the pressure-containing components of light-water reactors. This section is unique in recognizing the impracticability of direct access to radioactive areas and makes provisions for examination by remote means using techniques and equipment requiring development. Since there is no requirement for the performance of remote examinations until the end of the inspection period—or about 10 years from inception—allowance is made for changing of rules at some later time should developments so dictate.

The reactor-plant owner is clearly made responsible for the performance of the in-service examination including:

- —Development and preparation of written examination instructions and procedures.
- —Verification of the qualifications of personnel who perform the examinations.
- —Maintenance of adequate records of inspections performed.
- —Recording of all information and inspection results.
- —Retention of all inspection records.

In addition, the owner arranges for a duly authorized inspector to have access to all parts of the plant necessary for making the required inspections. The inspector must be employed by a state, municipality, or other enforcement authority having proper jurisdiction or an insurance company authorized to write boiler or pressure-vessel insurance. The qualifications of inspectors and inspection agencies are spelled out.

### Inspection Philosophy

The code is concerned only with components essential for the protection of the public health and safety. These involve principally those components of the reactor coolant system and portions of the auxiliary and emergency core-cooling systems connected to the reactor coolant system which constitute the reactor-coolant-pressure boundary.

The areas selected for examination are predominantly those associated with welds in the pressure-containing components. Why predominantly welds? The Code Committee determined that volumetric examination of welds includes a reasonable amount of the base material including the weld heat-affected zone. Thus welds are singled out as convenient locations to satisfy the requirements for both base and weld metal. Especially considered are dissimilar-metal welded joints, which are subject to more critical design and manufacturing control.

Two basic philosophies were involved in examination area selection:

1. Periodic inspections of areas having the highest service factor related to operating conditions. Examples are the weld seam between the reactor-vessel head and the flange, the inside-radius section of primary nozzles, and material affected by high neutron fluence.

2. Representative sampling to provide an assessment of the general overall condition. Examples are longitudinal and circumferential weld seams on the reactor vessel and piping, weld seams in pumps and valves, and bolting in the pressure-containing boundary.

A significant departure from the foregoing philosophies is allowed in the reactor-vessel region opposite the core. Although access to allow conduct of examinations is required, a reasonable period of time is allotted for development of remote inspection techniques. Similarly, no available technique can volumetrically inspect the welds between the austenitic-stainless-steel and high-nickel-alloy brackets and the interior vessel wall, but a more frequent and comprehensive visual examination is specified.

The code specifies that a base-line examination be conducted prior to initial operation and be as closely

representative of the examinations to be performed later as practicable. This can be performed either in the shop or in the field, providing it is performed after the hydrostatic test.

Evaluations of indications are to be related to acceptance standards involved in the original construction. Evaluation is then made to assess whether the condition indicated requires repair or other corrective actions, and, if not, disposition follows. In the words of the code:

> The intent of this Section of the Code is to provide for a review, on a case by case basis, of the indications to relate the condition to the service requirements to which the component is subjected. For the case where the review shows that no changes have occurred, disposition would require nothing further. For the case where the evaluation shows indications not previously detected, or larger than reported from previous examinations, but still within the original acceptance standards, disposition may require the examination of an additional number of components or like areas.
>
> Where the evaluation indicates that the condition does not meet the original acceptance standards, disposition requires additional examinations of this and like areas. It may be determined that the condition can be tolerated or that repair or replacement is necessary. In the event that repair or replacement is required, the intent of this Section of the Code is that restoration should be governed by the original acceptance standards.[*]

### Accessibility Requirements

The design and arrangement of the system components and piping must include allowance for adequate clearances to permit conduct of the examinations. Equally important, but not generally considered, are the handling equipment and storage space needed to allow temporary removal of structural members, shielding components, insulation, etc., to allow access to various parts of the system. Should defects or indications be revealed, consideration must be given to access for alternative examinations and/or repair or replacement, if so required.

### Scope

**Examination Techniques and Procedures.** Three examination categories are defined:

*Visual.* Visual examination remains the quickest and least-expensive method to determine the general surface conditions. Often evidence of misalignment or relative movement of mating parts can be quickly spotted. Direct examination is possible when access allows placing the eye within 24 in. at an angle no less than 30° to the surface. Where such access is not possible, remote means such as telescopes, periscopes, borescopes, fiber optics, or television cameras and monitoring systems can be used, provided they demonstrate the ability to provide resolution at least equivalent to direct means.

*Surface.* Surface examinations delineate or verify the presence of surface or near-surface cracks or discontinuities. Magnetic-particle and liquid-penetrant tests are provided for.

*Volumetric.* Volumetric examinations are intended to indicate the presence of subsurface discontinuities by methods capable of examining the entire volume of metal. Of the three categories of inspection, volumetric involves the most uncertainties. This is because the application techniques are yet to be fully developed for the geometries and remote locations involved. Two techniques are permitted:

- *Radiography*, using X-rays and gamma rays in accordance with Section III of the code.
- *Ultrasonics*, using the pulse–echo method in accordance with Section III of the code and with ASTM procedures.

Allowance is made for other ultrasonic procedures or newly developed techniques provided they are demonstrated to be at least equivalent to the pulse–echo method.

**Hydrostatic Tests.** A system hydrostatic test is required at or near the end of the inspection interval. On conclusion of the 4-hr test, examinations of the system are to be conducted. The type of examination is not defined, but the visual method is basic. The code does mention the developing technique of acoustic emission as important toward obtaining data on potential flaw growth at test pressure. The selection of test temperature is governed by three criteria: (1) fracture-toughness criteria for ferritic materials in system components; (2) brittle-fracture prevention criteria applicable to the reactor-vessel material as modified by the surveillance program during service lifetime; (3) system design and operational considerations related to test procedures.

Alternative rules are allowed for higher test pressures and different temperatures to validate the frac-

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ture resistance of a system component but are subject to appropriate review by the pertinent enforcement authority.

### Qualifications of Personnel for Nondestructive Examination

An examination is only as valid as the examiner is proficient in the techniques and procedures being used. Personnel performing nondestructive examinations must be qualified in accordance with American Society for Nondestructive Testing TC-1A wherever applicable. These documents do not cover all tests, and in such cases the owner must qualify personnel to comparable levels of competency. The inspector must verify operator certification and audit examinations.

### Examination Requirements

Preoperational Examination. All specified detailed examinations are to be performed once prior to initial plant startup. An important exception is that, in all cases, 100% of the pressure-containing welds are to be examined. Shop records are allowed on individual components and piping, provided equipment, techniques, working conditions, and recording methods are equivalent to those employed for subsequent in-service examinations. In actual practice this proviso may be hard to enforce since in-service inspection techniques have not really been developed. Of course, some base examinations are needed on the field welds.

**Examination Internals.** A 10-year cycle after commercial service (the inspection interval) is specified within which all required examinations are to be completed. With certain exceptions, 25% of the required examinations are to be completed by the end of one-third of the inspection interval and 50% by two-thirds.

The examination frequency, method of examination, and description of areas to be examined are described for 50 different components and parts are defined. For example: The volumetric method is required for the longitudinal and circumferential shell welds in the core region of the reactor vessel, including base metal that is one plate thickness beyond the edge of the weld. The code describes the extent and frequency as follows:

> The individual examinations may be performed at or near the end of each inspection interval and shall cover at least 10 percent of the length of each longitudinal weld and 5 percent of the length of each circumferential weld. In any case, for any individual area examined, the length of weld examined shall not be less than one wall thickness.
>
> When the longitudinal and circumferential welds have received an exposure to neutron fluence in excess of $10^{19}$ nvt ($E_n$ of 1 MeV or above), the length of weld in the high fluence region to be examined shall be increased to, at least, 50 percent.[*]

## REACTOR SYSTEMS

The code identifies those components in a reactor-coolant-pressure boundary which are essential for the protection of the public health and safety and which are thus subject to in-service inspection. Figure 1 shows the schematic of a typical PWR plant. In general, these components apply:

—Reactor vessel, head, and closure.
—Primary piping and nozzle attachments.
—Pump casing.
—Pressurizer.
—Portions of steam generator on primary side.

Direct-cycle boiling-water-reactor (BWR) systems do not have pressurizers or steam generators but do have more complicated internals.

For pressurized-water and boiling-water reactors, the inspection requirements apply to all pressure-containing components within the following system boundaries:

> (*a*) *The Reactor Coolant System.* For a nuclear power unit of the direct-cycle boiling-water type, the reactor coolant system extends to and includes the outermost containment isolation valves in the main steam and feedwater piping, which are capable of external actuation, and the reactor-coolant-system safety and relief valves.
>
> (*b*) *Portions of the Associated Auxiliary Systems Connected to the Reactor Coolant System.* For piping of these systems which penetrates primary reactor containment, the boundary extends to and includes the first containment isolation valve outside the containment capable of external actuation. For piping of these systems which contains two valves, both of which are normally closed during normal reactor operation, the boundary extends to and includes the second of these valves (the second of which must be capable of external actuation), whether or not the system piping penetrates primary reactor containment.
>
> (*c*) *Portions of Emergency Core-Cooling Systems Connected to the Reactor Coolant System.* For piping of this system which penetrates primary reactor containment, the boundary extends to and includes the first containment isolation valve outside containment capable of external actuation. For piping of this system which does not

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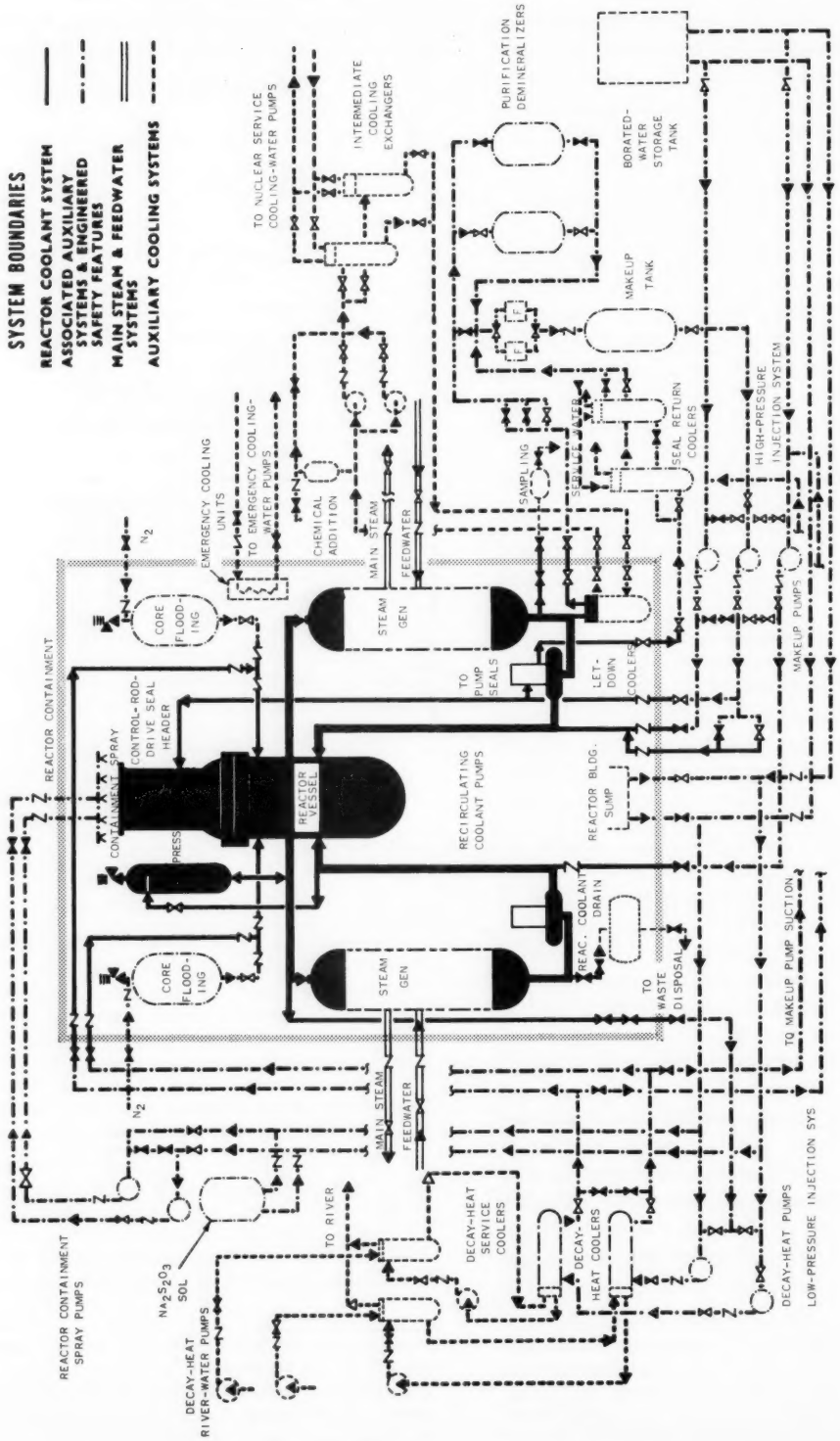


Fig. 1 Typical PWR plant (Fig. B in Section XI of the ASME Code.)

penetrate primary reactor containment, the boundary extends to and includes the second of two valves normally closed during normal reactor operations.

(*d*) *Excluded from examination* are those components and piping, within the boundaries of (*a*), (*b*), and (*c*), where postulated failure would not result in loss of reactor coolant that exceeds the capability of normal makeup systems for the interval of time required to permit a reactor shutdown and orderly cooldown from the respective conditions of startup, hot standby, operation, and cooldown.[*]

Revisions to this boundary are pending.

## IN-SERVICE INSPECTION METHODS

The objective of in-service inspection of nuclear power-generating systems is to ensure the continued physical integrity of the pressure boundaries. This objective is to be accomplished by the acquisition, processing, and evaluation of pertinent information by nondestructive-testing methods. The critical information aspects of this process are the detection of significant discontinuities, the unambiguous processing of the inspection signals, and the proper evaluation of a sufficient amount of data to reach a correct decision.

Before the details of an inspection process can be formulated, it is necessary to form an inspection philosophy. Unfortunately this is not a simple matter. The proposed code for in-service inspection is based on the present manufacturing codes. The manufacturing codes are, in turn, based on standards that are compatible with manufacturing capabilities, nondestructive testing methods, and a wealth of experience from power plants. That is, the criteria for defining objectionable discontinuities are based on the ability of present methods to reduce or eliminate discontinuities and not on the effect that these discontinuities would have on performance. Some of the nondestructive-testing inspection methods used in manufacturing will not be applicable to in-service inspection due to inaccessibility and radioactivity. In many cases the in-service inspection would be made by a different method or procedure, and the inspection results could not be directly compared. Because inspection results used in manufacturing are of little use for in-service inspection comparison, it is necessary to conduct a preoperational base-line inspection, either at the fabricating facility or on site, with suitable records, using in-service inspection procedures. Once this base line is established, it serves as the basis for comparison of subsequent inspections.

In-service inspection is primarily a search for cracks that might lead to fracture. Cracks tend to be mirror-type reflectors; detection and evaluation of crack-type discontinuities are more reliable than diffuse-type discontinuities. The definition of a "defect" may be limited to crack-type discontinuities, and inspection procedures should be optimized for this type of discontinuity.

The inspection methods used for direct examination, such as by visual and surface methods, are not described here since they are well documented in ASME and American Society for Testing and Materials (ASTM) literature. A good summary of the various inspection methods is contained[4] in Report WARD-3977-1. Although the document is primarily concerned with the needs of the Fast Test Reactor, the authors have made a comprehensive survey of the state of the art as applied to all reactor systems.

Diagnostic tools using noise analysis, acoustic viewing, and vibration analysis are described. These techniques are in their infancy as applied to reactor systems but are expected to find use for monitoring purposes. It must be pointed out, however, that such devices do not qualify as methods for in-service inspection as defined in the code.

This article will concentrate on the techniques being used and developed for inspection of reactor vessels with the attendant high radiation levels and complex uniform and nonuniform shapes. This technology, although relatively new, is developing rapidly.

### Environment for Examination and Criteria for Methods

Figure 2 is a representative PWR reactor vessel of Babcock & Wilcox design. The reactor vessel may be considered as having four individual components, each with its own inspection problems:

1. *Nozzles.* The primary inlet and outlet nozzles and the core-flooding nozzles are particularly difficult to inspect owing to their shape and size and the inaccessibility of their inner surface. The inlet nozzle has the simplest shape of the three. The outlet nozzle presents an additional difficulty in the buildup of material at its knuckle (for flow-direction purposes), which complicates the knuckle inspection. The core-flooding nozzles contain thermal sleeves, which completely obviate inspection from their interiors.

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2. *Straight cylindrical section.* This section of the reactor vessel, which constitutes the bulk of the vessel and receives the greatest neutron dosage, is perhaps the most straightforward to inspect.

3. *Instrumentation penetrations.* These penetrations are on closure and bottom heads and on control-rod-drive nozzles on the closure head.

4. *Bottom-head weldments.* These weldments are a region of variable cross-sectional dimension and nonuniform weld configurations.

A BWR vessel is somewhat more complicated in that the control-rod drives are located in the bottom head, and there are more thermal sleeves.

The principal characteristics for the ideal remote inspection method are that it:

1. Be nondestructive.
2. Provide appropriate sensitivity resolution.
3. Allow automatic, remote operation.
4. Be insensitive to background gamma rays.
5. Allow in-depth volumetric inspection.
6. Allow one-surface access.
7. Provide data output capable of being electronically processed.
8. Provide data capable of being stored and retrieved.
9. Give precise data from inspection to inspection.
10. Allow rapid operation and analysis.

The gamma-radiation level outside the pressure vessel is on the order of 10 R/hr at shutdown. This level is a biological hazard that precludes the legal presence of an operator in the area for more than a relatively brief period of time. The inspection method must allow inspection in depth, for the entire volume at the region of interest, and must be surveyed to determine fully the presence of discontinuities and to chart their propagation.

Further, the space usually available around a reactor vessel is quite limited and certainly not sufficient to allow the movement needed by a human operator. The other possibility is examination from the inside after removal of the core and internals. Although more space is gained, the background radiation is still too high for use of direct means.

Candidate nondestructive-inspection procedures include magnetic-particle, dye-penetrant, eddy-current, radiographic, and ultrasonic methods.

### Surface-Discontinuity Indicators[5]

Magnetic-particle and dye-penetrant methods for detecting discontinuities are difficult to adapt for remote applications. The surface must be cleaned, and the indicator must be applied and then removed remotely. Crud buildup would be an interference, and with present techniques the indicators could not be used in submerged areas. The gamma background and the vessel temperature might cause deterioration of the indicator components. Surface-discontinuity indicators are limited to flaws at or close to the surface. They cannot determine the depth of a defect. Magnetic-particle inspection requires the induction of a magnetic field into the steel. If this is done by electrode contact, there is a possibility of arcing, which would create a local source of crack initiation, and this would not be acceptable.

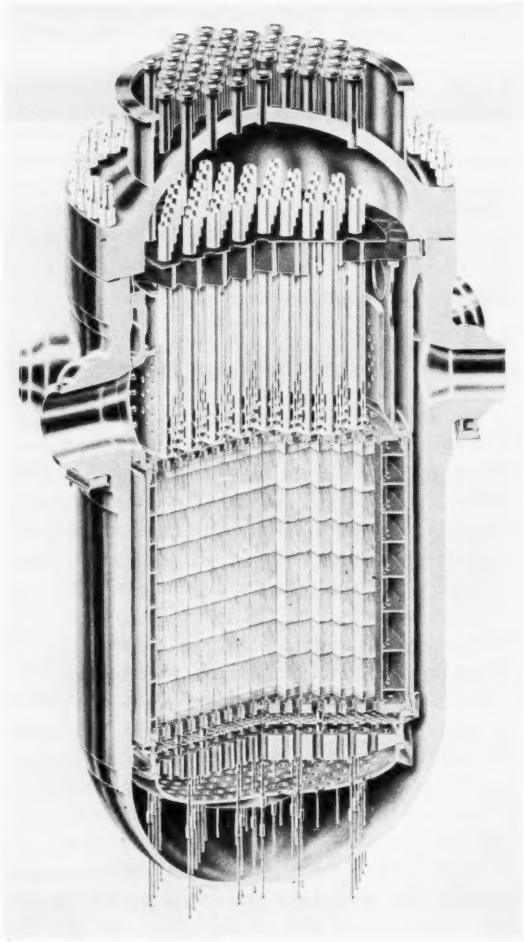

Fig. 2 Typical PWR reactor vessel of Babcock & Wilcox design.

### Radiographic Inspection[5,6]

In radiography an X-ray or gamma-ray beam is directed through the material to be examined and then onto a receptor, such as a photographic film or a fluorescent screen. Materials of different density will attenuate the beam differently; high-density steel is of relatively greater opacity than the lower density material included in a crack or slag inclusion. In thick sections of steel, it is difficult to detect tight cracks by radiography, unless the defect were oriented normal to the X-ray beam. A quantitative description of the orientation and the size of a discontinuity would depend to a large extent on the actual conditions involved. Resolution usually requires obtaining radiographs from a number of angles.

Background gamma radiation from the neutron-activated steel interferes with the purposely introduced "inspection" radiation. This interference requires such measures as the placement of lead plates between the vessel wall and the receptor. Besides being inconvenient, higher intensity radiation sources would be required to override this shielding. Such equipment is large, expensive, and difficult to handle in the limited space of a reactor containment. Data storage is an additional problem. The output of radiographic inspection is a pictorial representation, requiring the filing of many shadowgraphs. Radiography requires access to both sides of the vessel wall since the X-ray or gamma-ray source is on one side and the receptor is on the other. Such access is not always practical and, in many instances, is not possible.

### Ultrasonic Inspection

In ultrasonic inspection[5] a directional sound wave of high frequency, about 1 to 10 MHz, is introduced into the test piece, and its reflection from, or attenuation by, a flaw in the material is detected. The beam is directed through the material to provide information on the condition of material through which the beam passes. The sensitivity to a discontinuity is dependent on many factors, including spectral output of the transducer and the nature of the reflecting surface. The tightness of a crack is usually unimportant, except in cases of extreme tightness where defect recognition is reduced. Generally, ultrasonic testing is much more sensitive in this respect than radiography. A discontinuity perfectly aligned with the beam direction would probably go undetected. Sonic impulses applied from more than one direction are used to detect discontinuity aligned with single beam direction.

Steel is not a perfectly homogeneous material because of variations in the inclusion content and the grain size. Also, the surfaces of the vessel are not perfectly regular. All of these factors can irregularly attenuate or interrupt the ultrasonic beam. This causes a variation in the oscilloscope signal's peak heights from sound material. The ASTM methods recognize this characteristic by requiring calibration of signal amplitudes using special test blocks of the material to be tested.

There are three general ultrasonic inspection methods currently available: pulse–echo, through-transmission (including transmit–receive), and "delta."

Work on visualization procedures shows great promise, but more application development is needed. The current techniques of $X$–$Y$ plot and $c$-scan recording must be shown to be applicable to the quantity of data required and recall of information.

Ultrasonic inspection is strongly affected by component geometry. The complicated three-dimensional geometry of the reactor vessel presents many difficult inspection problems. Hence the reactor pressure vessel is divided into a number of smaller inspection regions and each region is treated independently. The task of developing a successful universal inspection procedure applicable to all regions of the vessel is very difficult and has not been accomplished to date.

The necessary attributes that make ultrasonic inspection adaptable to remote means, as demonstrated by Babcock & Wilcox[7,8] and Southwest Research Institute,[9,10] are that it: allows inspection with access to only one side, is insensitive to gamma radiation, is capable of volumetric inspection of thick steel plate, is capable of remote operation with the output signal digitalized, has relative rapidity of performance, and has sensitivity to acceptable defects. However, the actual application is a formidable task. Working hardware is being installed in reactors as development continues apace to perfect techniques and equipment.

The inspection system must be completely automated and capable of traversing a rather large structure. Precise correlation of data from one spot from inspection to inspection is essential, for in that way the extent of propagation of a defect in an inspection interval can be determined. The data output will be very large because of the size of the components. The output must lend itself to automatic acquisition and storage either by photographic or digital means in order to record completely objective data and to conserve time.

**Through-Transmission.** In a paper[7] before the Annual Meeting of the American Society for Nondestructive Testing in October 1969, authors Gross and Johnson described the results of a program to develop an automated ultrasonic method for inspecting straight cylindrical vessel sections. The work was sponsored jointly by Babcock & Wilcox and Empire State Atomic Development Associates. After evaluation of various inspection methods, they chose the ultrasonic two-transducer, transmit–receive method, commonly called the "pitch-and-catch" technique. The concept is called Rapidscan (Remote Automated Pressure-Vessel Inspection Device). Its essential feature is the detection of changes in defect condition through changes in signal amplitude with minimal operator influence.

Figures 3 and 4 show the method and the appearance of the signal in regions both with and without defects. The beam is transmitted from one transducer to a second transducer. A defect-free material would result in an oscilloscope indication of maximum amplitude. However, if the beam path impinges on a defect, some of the ultrasonic energy is deflected, resulting in a decreased signal amplitude. Further beam impingement of the defect can result in a complete loss of the transmission signal. The process repeats itself in mirror-image as the beam passes over the defect. Careful analysis of the data allows determination of the length, depth, orientation, location, and direction of growth of a defect with good accuracy.

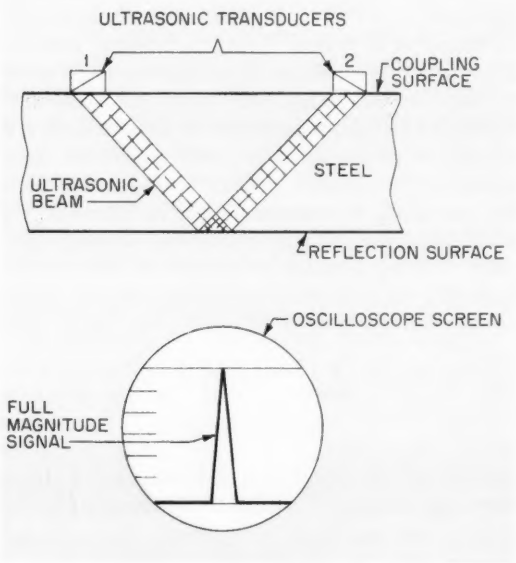


**Fig. 3 Ultrasonic inspection of a defect-free region using transmit–receive method.**

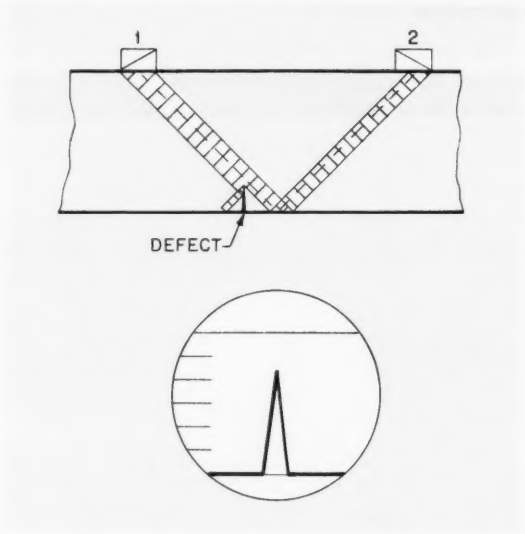


**Fig. 4 Ultrasonic inspection of a region with a "small" defect using transmit–receive method.**

Sperry-wheel transducers are positioned side by side in fixed geometry on the vessel's surface and communicate via reflection of the ultrasonic signal from the far surface. A Sperry wheel is a fluid-filled elastomeric tire enclosing piezoelectric crystals, which maintain a fixed attitude as the wheel rolls on the work surface. The Rapidscan system utilizing this device is shown in Fig. 5. A V-shaped beam is produced, which is transported through the entire volume of the vessel. The transducers, being in constant communication, afford a continuous signal, certifying that the inspection is being performed. A closed-circuit television camera accompanies the ultrasonic transducers to allow simultaneous visual inspection. The television camera and the two transducers are mounted on a rigid carriage that moves on tracks, a gear, or some other suitable mechanical device to scan vertically from bottom to top, index over circumferentially, and again scan vertically, this time from top to bottom, continuing in this manner until the entire volume of the cylinder has been inspected.

The concept was demonstrated on a test cylinder 4 ft high and 4 ft in diameter, with a 4-in.-thick wall. Test zones were incorporated, one having an as-rolled condition and three having stainless-steel weld cladding of varying surface roughness. Artificial defects of

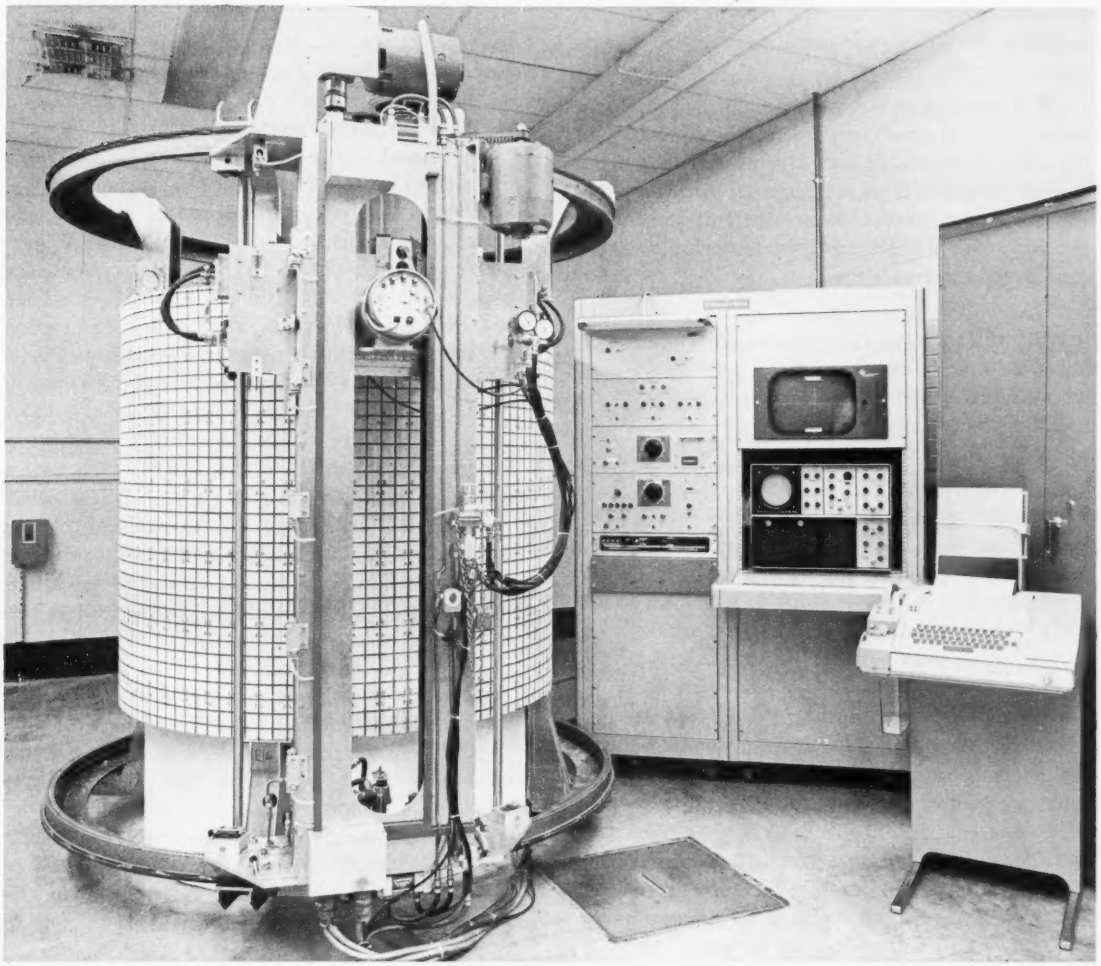

Fig. 5 Rapidscan system.

varying depth and width were introduced on the inner surface in the course of the demonstration.

The cylindrical (belt-line) section was chosen initially because this region receives the highest neutron dosage and is therefore the most likely to experience degradation of mechanical properties. In practice, this section contains several circumferential and longitudinal welded seams where the courses are joined. Steel thickness generally varies from 4 to 10 in., depending upon the particular reactor design.

By moving the transducer pair over the surface in a square wave pattern, the examiner can determine the length, depth, and orientation of defects. Comparison of signals in the predefect (base-line) and defective (postoperation) conditions using a "difference chart" of pulse-amplitude integers reveals the defect dimension. Defects as small as 0.1 in. deep and $\frac{3}{4}$ in. long may be detected and characterized completely automatically. Data output is seen on an oscilloscope screen as conventional ultrasonic "blips" and also is digitized for presentation to a teletypewriter and ultimately to a data-processing computer.

The preceding assumes a defect orientation that is optimum relative to the ultrasonic beam path. Capability of positively detecting defects of all orientations requires an additional transducer pair located 90° to the first. The detection of laminar defects with the ability to describe their locations and extent with

certainty requires use of a fifth transducer positioned normal to the surface and operating in the pulse–echo mode.

If a comparison is to be made with subsequent inspections, the signal received from a particular spot (or address) on the vessel wall must be referenced precisely to that address. A transporter mechanism that will guarantee address precision is therefore an absolute necessity. This was accomplished through a precision carriage and television system.

The work was conducted primarily from the outside of the vessel for experimental convenience. There is no reason to believe that inspection from the inner surface would not be equally suitable, as evidenced by a static experiment.

In Ref. 11, Gross describes the application of the transmit–receive technique to the more difficult inspection of nonuniform sections like nozzle-to-shell welds. Two transducers were positioned in fixed relation with each other and rotated about the outside vessel surface at the nozzle juncture. With a wide-angle (78°) shear wave, artificially induced cracks were detected in the most probable place for fatigue defects, e.g., inner diameter juncture of nozzle and vessel.

The author has described a remote inspection system featuring required sensitivity, reproducibility, and adaptability to electronic processing—all with minimal operator subjectivity. Some questions remain to be resolved, however. The ability of Rapidscan to characterize defects reliably remains to be proven. One question is the effect of surface finish upon beam dispersion. Inner-surface cladding, weld-deposited by the single-electrode process, has a wavy finish that disperses the ultrasonic energy rather than reflecting it in a relatively narrow beam. This effect has not yet been fully investigated but can be avoided by surface preparation or cladding by other processes yielding a smoother finish. Other problems include the unknown effect of crud buildup, miniaturization, and optimization of the ultrasonic-transducer material selection and size. Finally, qualification of this concept would probably require approval of a special code case before application.

**Pulse–Echo.** In the pulse–echo method, the piezoelectric transducer used to send the sound waves also receives the return signal. This approach allows alternatives in method and technique, all dependent upon a signal's being reflected back to the transducer from a defect. Special scanning techniques and angles of transmission are used to detect defects of varying orientations. Careful selection of methods can fully characterize a defect. Data acquisition and storage are generally by photographic means. Pulse–echo signals can be digitalized, but not easily, and interpretation of the digital output presents an additional difficulty.

Southwest Research Institute uses the pulse–echo method for its extensive inspection program. Various-angle axial and circular shear waves are used, plus automated scanning systems. In Ref. 9, R. D. Wylie of Southwest Research Institute describes a systematic program for surveillance of the La Crosse Boiling Water Reactor. The article is a brief survey of proposed methods for monitoring temperature, monitoring vessel irradiation level, and in-service inspection. Written in 1968, the article predates Section XI of the ASME Code. Rather, the author concentrates on ways to inspect critical areas of known geometric stress concentration in the vessel and piping system. These include the flange and nozzle welds to the vessel and head and nozzle-to-piping welds. Figure 6 shows the proposed scanning system for the head-to-forging weld. Four transducers emitting beams at varying shear-wave angles are used to cover three zones of the weld. In like manner, a system for inspecting nozzle-to-shell welds from the inside is shown in Fig. 7, as used for the Elk River reactor. A combination of transducers for angular and perpendicular shear waves and longitudinal waves is rotated in a probe. The actual configuration of a reactor nozzle is much more complex than that shown here. Fatigue-stress-induced defects can occur at planes of possible weakness such as the 6 and 12 o'clock positions of the inner-diameter junction between the nozzle and the vessel in a plane radial to both. Such a crack would be detected in the Southwest Research method with the 75° shear wave, relying on reflection from the multiple facets of a natural crack and utilization of Rayleigh waves. The Wylie article does not describe how well the technique locates and determines the size and orientation of such fatigue cracks.

Hawkesworth, Lautzenheiser, and Reinhart[10] describe the various methods and types of equipment used by Southwest Research for in-service inspection beginning in 1966 with the Elk River reactor vessel. An unexpected problem encountered has been that sometimes the "as-built" component drawings often are in error. In the Elk River reactor, vessel internals were different, nozzle bores were not round, and in one case the nozzle bore was not oriented normal to the inside diameter of the vessel. Thus laboratory-tested devices did not work well in the field without modification. It should be noted here that many current reactors were

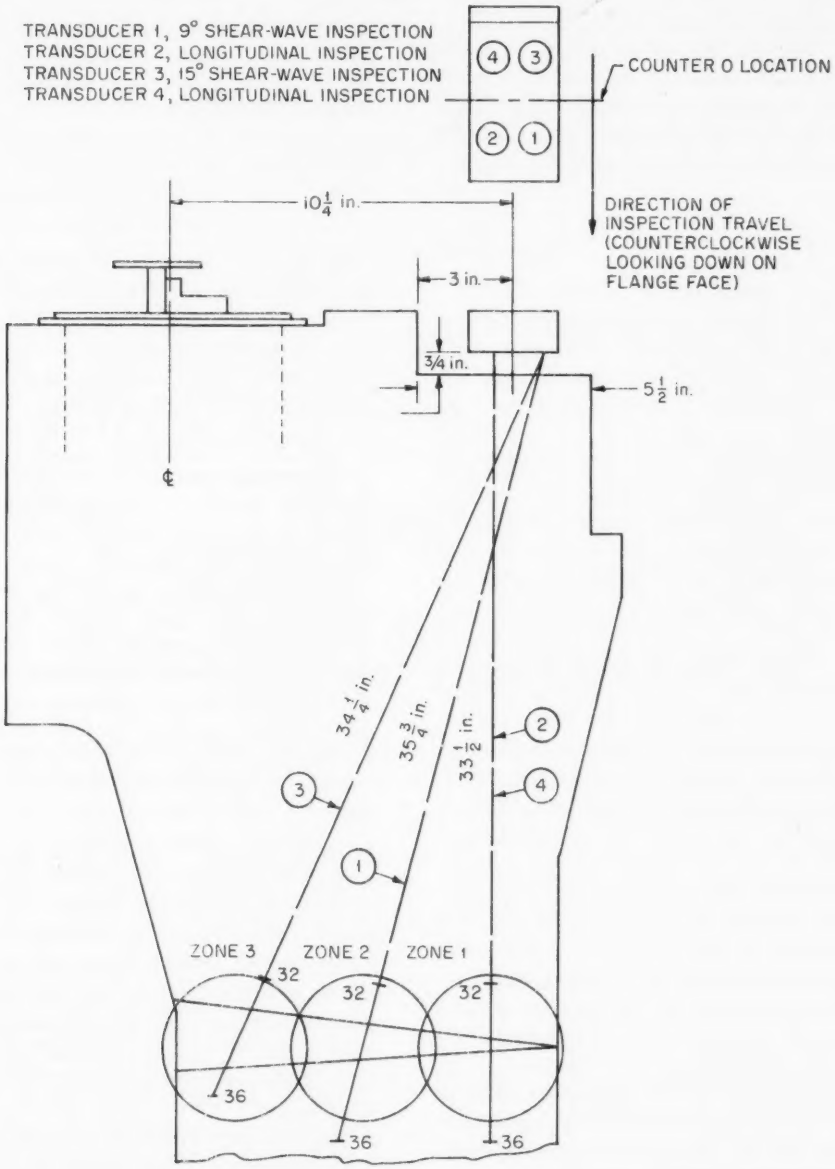


Fig. 6 Ultrasonic scanning system for 11 head-to-forging welds, Southwest Research Institute.

designed before the need for in-service inspection was defined, and adequate clearances are not provided.

The data-acquisition approach of Southwest Research centers on the use of a stop-motion movie camera to photograph a panel containing meters and counters showing the linear position of the examination device and the position of the transducer modules relative to the center of the propulsion device and their angle. A placard displays the nomenclature of the weld being examined, the cross section of the weldment, and the method of examination. Simultaneously, a strip-chart recorder indicates the position and amplitude of the ultrasonic signal along with linear position information for correlation with the photographic record.

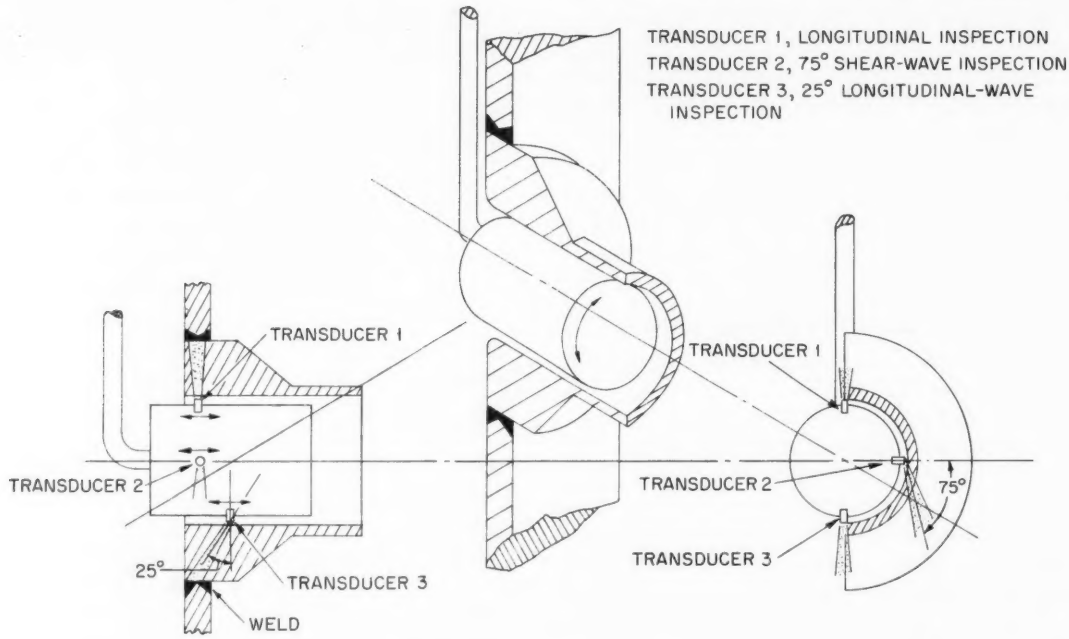


**Fig. 7 Ultrasonic scanning system for 11 nozzle-to-shell welds, Southwest Research Institute.**

Southwest Research has designed and used several specialized inspection devices. The BWID (Butt-Weld Inspection Device) will inspect welds in piping 10 in. or greater in diameter. The device is a propulsion unit with an adjustable three-transducer module on each side. Two of the transducers are variable angle and thus have the same impingement point on the pipe surface when the angle is changed. The other transducer is fixed for longitudinal examination and monitoring of the water cavity and transducer coupling. This arrangement allows the examiner to use pulse–echo, pitch-and-catch, or any combination.

The Programmed Linear Ultrasonic System (PLUS) is a device used primarily for examining longitudinal and circumferential welds of pressurizers and steam generators. The distance of the transducer module from the propulsion device is adjustable. Operation can be programmed for straight or oscillatory motion.

A special device for remotely inspecting nozzle interiors has been designed to simultaneously examine safe-end welds and nozzle-to-vessel welds and inner radii. Either the reactor or the refueling crane is used for manipulation, with firm support provided at the stud holes.

All developments at Southwest Research benefit from valuable early experience. However, as with all systems, the ability to meet the intent of Section XI remains to be shown in practice over the next few years.

**Delta Method.**[9,13] Automation Industries is developing a procedure whereby an inspected area is irradiated with ultrasonic energy, and the reradiated energy from a defect is detected. At least two transducers are required: one introduces the ultrasonic energy into the material, and the other is focused directly over the inspected area receiving the signal from a defect. Theoretically this procedure would enable high-speed inspection for a defect regardless of its orientation.

The method is not completely characterized and is not in general use. Developmental work is under way.

**VUER.** Selig, Kosky, and Hedden[13] describe an in-service inspection system developed by Combustion Engineering, Inc. The heart of the system is VUER, a device designed to perform visual and ultrasonic examinations simultaneously. The VUER is transported to the inspection area by the refueling machine and is hung from that machine while the inside diameter of a vessel is being inspected. Pulse–echo techniques are used, with transducers mounted on hydraulic cylinders. Visual inspection is by television.

Calibration and repeatability of positioning are proposed to be accomplished by the use of an indexing marker system, such as a weld buildup accurately near the intersection of girth and longitudinal welds on the outside diameter of a vessel.

This article does not describe any actual performance or application of this system, but, rather, a general approach to in-service inspection.

**SKATE.** Hawkesworth, Lautzenheiser, and Reinhart[10] briefly describe a concept of Atomics International for use in examining sodium-cooled reactors. As in VUER, the system employs an ultrasonic module and television camera for performing simultaneous ultrasonic and visual examinations. However, the device apparently has not yet been used in the field.

**Tatzelwurm.**[10] This device was developed by Dr. Ing. Hans Meyer at Maschinenfabrik Augsburg-Nuremberg Aktiengesellschaft, Nuremberg, Krautkramer Gesellschaft für Elektrophysik, Cologne, and Institut für Reactorsichheit der Technischen Überwaehungs-Vereine e.V., Cologne, under contract with the Federal Republic of Germany for shop use on the Würgassen BWR reactor vessel. This equipment will be used for in-service inspection of the Kernkraftwerk Obrigheim Siemens PWR at Obrigheim, Germany. The device consists of an array of pitch-and-catch and pulse–echo sensors that sequentially project a series of precisely positioned ultrasonic beams into a component to detect through-the-thickness cracking anywhere in the wall thickness. Eight to 10 probes, each having two or more transducers, are hinged together so an articulated assembly can conform reasonably well to the curvature of the work surface. The reference describes in some detail how the device works. Tatzelwürm has been demonstrated in the laboratory on a model of a clad vessel wall but apparently has not been on an actual vessel to date. Although the system offers the advantages of detecting vertical flaws and better uniformity of signal, its complexity introduces high cost and weight as well as possible ineffectiveness where obstructions exist on the component surface.

## IN-SERVICE INSPECTION OF OPERATING REACTORS

The reader attempting to get detailed information on early inspection development through the literature will be disappointed. The companies in the field who provide such services to utilities are only now beginning to release information. On the other hand, the actual inspections of operating reactors are documented, often in considerable detail, in the mandatory reports issued under license provision to the AEC Division of Reactor Licensing. These reports, plus various papers, reveal an increasing sophistication in the art. Much of the experience gained was used to establish the present ASME Section XI In-service Inspection Code.

### Yankee Rowe[4]

This 175-MW(e) pressurized-water reactor (PWR) by Yankee Atomic Electric Company, located at Rowe, Mass., went into commercial operation in 1961. A comprehensive repair operation was conducted in 1965 as a result of the loosening of some internals with subsequent damage to other parts. This repair coincided with Core IV–Core V refueling operation.

Inspection played an important part in determining the extent and cause of some failures, ascertaining that some critical components were sound, and determining that the repair operations were successfully completed. Most inspections were conducted through water up to 40 ft deep. In some cases contact inspection out of water was possible using a combination of temporary shielding and rapid work to limit radiation exposure. The inspection techniques were:

1. *Visual Inspection.* A periscope and a borescope were used to view many of the internals components, including all bolts, radial bearing pads, and new components on the thermal shield following installation.

2. *LVDT.* A linear-variable-differential transformer (LVDT) was used to measure the wear on the radial bearing pads.

3. *Replication.* Silicone rubber (G.E. RTV-88) was cured in place to obtain replicas at several locations. These included the wear pattern on the radial bearing pads, a wear pattern in the vessel-bottom-head cladding and relative displacement of sections of the thermal shield (top and bottom).

4. *Dimension Measurements.* Measurements in the vertical direction were made with a tape attached to a long pole. The bottom of the pole was placed against a surface on the internals, and the tape was read with a leveled transit. This reading was compared with readings on other surfaces. Radial measurements were made, first, by measuring the vertical movement of a wedge between the parts being measured using the tape and transit, and, second, by using a plumb bob on the end of an insulated wire. The position of the plumb bob was measured with a scale at the top of the wire. An electric circuit consisting of the wire, the internals,

and an ohmmeter was used to indicate contact of the plumb bob with the internals.

5. *Barrel-Flange-to-Gusset Welds.* The integrity of these welds was assured by using dye-penetrant, ultrasonic, and visual examinations. This inspection was done with the welds out of water.

The internals were reinspected using the same methods during the Core VI–Core VII refueling operation in 1968 and found to be in good condition. Many of the same techniques were used. In addition, a closed-circuit television system was used for visual examination.

### Dresden 1

Inspections of Commonwealth Edison Company's 200-MW(e) boiling-water reactor (BWR) at the Dresden Nuclear Power Plant, Unit 1, are among the most well documented in the industry. More than the usual detail will be given here because of the wealth of field information provided. Three comprehensive reports[14-16] describe inspections conducted at the spring 1967 refueling outage. From 1960 to 1967 Commonwealth Edison had made routine inspections consisting in the usual observations of important and base-metal welds conditions and ultrasonic testing of closure studs. During these inspections, 23 major welds in the primary system were uncovered and examined. These examinations gave valuable experience in inspection procedures in radiation areas, and this was helpful in planning and carrying out the 1967 inspection program.

By 1967 Dresden 1 had the longest operational experience of any large power reactor in the country. In view of that history and because, in the 1966 inspection, cracks had been found in small-diameter seamless-steel pipe in the primary-system recirculation-valve bypass lines, Commonwealth Edison decided to give the reactor pressure vessel and its associated piping an extensive inspection. The utility contracted with Babcock & Wilcox and General Electric to conduct certain programs, using the latest nondestructive-testing techniques. The Babcock & Wilcox tests included the reactor-vessel closure head, inlet nozzles, and flange, and the steam-generator risers and downcomers. General Electric inspected the welds in the stainless-steel piping and at the control-rod-drive housings.

Reference 14 includes three reports. The first by Commonwealth Edison describes the inspection planning, the tests carried out by its own personnel, and the support it gave the two contractors. The Babcock & Wilcox and General Electric reports describe the testing techniques used and the results obtained, and each draws some conclusions from the program.

Figure 8 shows schematically the arrangement of the Dresden 1 primary system. The reactor is of the dual-cycle type. (Later BWRs eliminate the secondary steam generator.) The 40-ft-high, 12-ft-ID reactor pressure vessel is made of $5\frac{5}{8}$-in. thick A302-B steel plate roll-clad on the inside with $\frac{3}{8}$-in. 304 stainless steel. All the many nozzle connections in the vessel were fabricated in the same manner: a field weld of 304 stainless steel to 304 stainless steel to the piping with a bimetallic shop weld of 304 stainless steel to A302-B. Table 1 summarizes[16] the scope of the 1967 inspection.

Holyoak[14] makes a number of important observations about the need for careful advanced planning and the realities of handling very heavy radioactive pieces, like the reactor-vessel head. The inspection had to be fitted into the fueling outage to minimize downtime. This required coordination and detailed drawings of test and maintenance operations, including isometrics to aid in locating and recording inspection areas. Although inspection of the control-rod thimble welds at the bottom of the vessel required less than 4 hr each, the inspection team had to wait for the maintenance and operating forces to remove the control-rod drives. This involved removing the 8-ft control-rod-guide tube and sealing the top of the thimbles with a special tool, working through 50 ft of water, before the drive itself could be removed. Only one drive could be removed at a time, with a several-day cycle between drives. Radiation levels in the thimble areas were about 50 mR/hr. Inspection of steam-generator loops was not difficult. The average radiation level here was 500 mR/hr.

The reactor-vessel flanges and closure head were another story. In the latter case a magnetic-particle check of the top and a dye-penetrant check of the underside cladding were performed. The usual horizontal position used for storage during refueling was a good position for the top-surface inspection. It took a week to reposition the 50-ton head to a vertical position for the underside inspection. Only a single-hook crane was available, and cladding extended halfway into the stud holes that had to be used to sling the head in the new position, thus complicating inspection. Final positioning required four 16-ton chain falls, which, in turn, required moving a gallery, cutting holes in cement floors above for access to

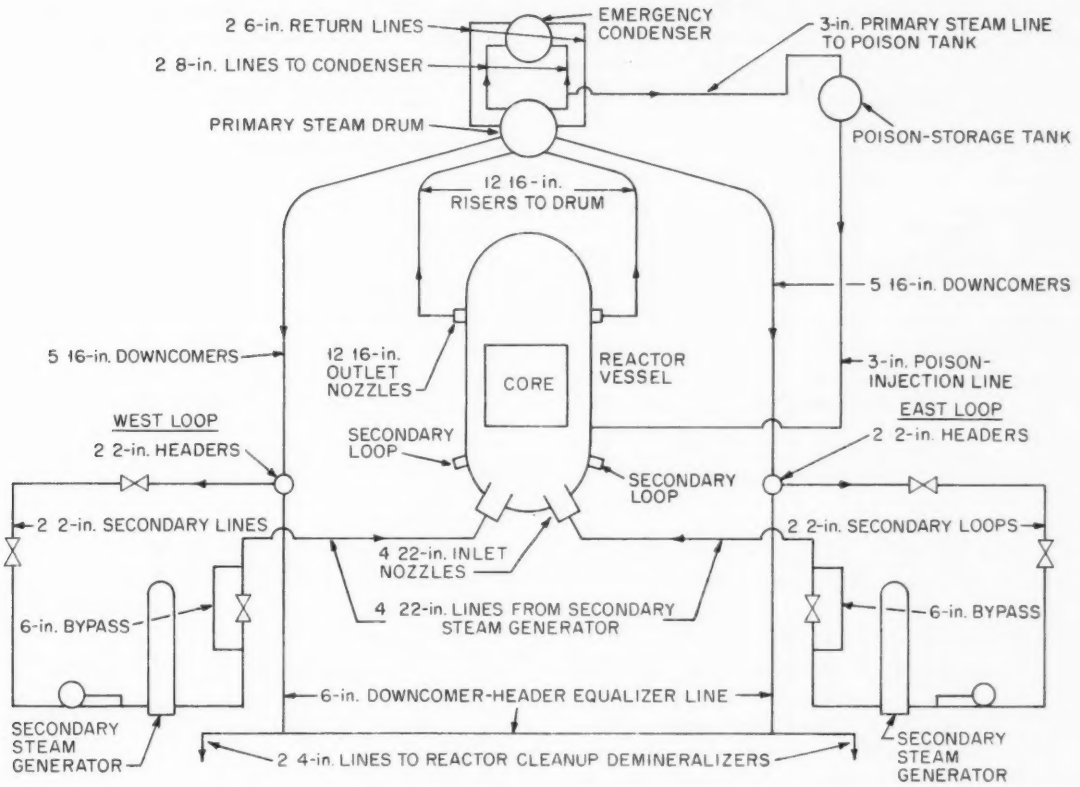


**Fig. 8 Schematic drawing of Dresden 1.**

beams, and providing a support cradle on which to rest the head in its new vertical position on the floor.

Another week was required to decontaminate the head. Radiation levels 6 ft from the inside surface were 6 R/hr. This work was done while fuel-movement work was being carried on in the containment sphere. The need for face masks by men working in the sphere was eliminated because the working area was enclosed in a plastic tent that prevented airborne radiation. Very large trays were set down to collect the decontamination solutions, along with pumps and a tank in which to store the used radioactive decontamination solutions, until sphere integrity could be broken and the solutions could be pumped out to be solidified for shipment and burial.

Radiation levels were reduced to workable levels by a combination of jet washing, chemical coating and washing, and scrubbing with long-handled brushes, followed by steam and hot-water cleaning to remove any oils and nickel oxide film that would interfere. The metal, originally coated with brown corrosion products, cleaned to a bright condition. Since the cladding was originally machined, no grinding was necessary.

Examination of the bimetallic welds at the inlet nozzles involved problems of access and contamination. A 4-ft-thick block shield wall had to be removed at each of the four inspection locations to allow access to the welds. The location of these welds required the use of ladders in some cases and working within a restricting insulation holding ring that covered part of the shop welds on these pipes. The work on these welds and the reactor inlet welds required extensive insulation removal. This caused a later problem of boxing all the contaminated insulation waste for radioactive-material burial. Gamma radiation levels in the inspection area were about 2 R/hr. Wet contamination required inspectors to wear half-masks and rubber clothing.

Forrer[14,15] has documented the techniques used in the inspection of the closure head, vessel inlet nozzles, vessel flange, and steam-generator risers and downcomers.

Table 1 Summary of Dresden 1 Inspection During Outage Beginning Jan. 13, 1967*

| Item | Inspection | | Anomalies | Action |
|---|---|---|---|---|
| | Extent, % | Method | | |
| Control-rod drives | 14 | Visual; dye check | None | Overhauled |
| Fuel | | | | |
| Channels | 100 | Visual | 2 | Not safety significant |
| Orifices | 100 | Visual | Fouling | Cleaned and flow checked |
| Assemblies | 100 | Visual; 63% flow checked; 92% sipped | 25 leakers† | Removed |
| Primary system | | | | |
| Reactor head top surface | 100 | Mag-Part‡ | None | None required |
| Reactor head inner cladding | 100 | Dye check | 28 indications | Not safety significant |
| Reactor flange weld | 100 | UT§ | None | None required |
| Reactor flange surface | 100 | 100% dye check; 27% UT | None | None required |
| Control-rod-drive thimbles and J-welds | 14 | UT | None | None required |
| Large-pipe welds | 15 | UT; dye check | None | None required |
| Small-pipe welds¶ | | | | |
| 10 in. | 83¶ | UT | None | None required |
| 8 in. | 100¶ | UT | None | None required |
| 6 in. | 100¶ | UT | 12 | Replaced and repaired |
| 4 in. | 100¶ | UT | 1 | Replaced |
| 3 in. | 94¶ | UT | 3 | Repaired |
| 2 in. | 50¶ | UT | None | None required |

*From Ref. 16.
†Type I, 19; type IIIB, 2; and type IIIF, 4.
‡Magnetic particle.
§Ultrasonic testing.
¶To first valve that can be closed for damage control.

**Closure Head.** The top surface was magnetic-particle tested using a Magnaflux Corporation type "Y" yoke with 110-V power source and 5-in. prod spacing. At each location the prods were rotated 90° to assure detection of discontinuities in all directions. None was found. Movement of the head was required for downhand inspection. The dye-penetrant examination of the underside incorporated an innovation to reduce personnel radiological exposure. A more rapid test was achieved through the use of postemulsified, water-washable dye-penetrant materials, which were applied from aerosol cans. Not only was the exposure reduced but also the amount of radioactive waste from the inspection operation. A number of small, random indications were observed and judged to be harmless.

**Reactor Nozzles.** Dye-penetrant inspection using nonwater-washable materials revealed no unacceptable defects. Ultrasonic testing was performed using Sperry Models UR-600 and UM-721 equipment. The couplant for all tests was "Hamikleer." Test weldments representing the welds to be examined were used for calibration of ultrasonic equipment. The weldments contained ¼-in.-diameter flat-bottom holes drilled ½ in. deep for calibration of the normal-wave transducer. In addition, notches of 60° V configuration and of depth equal to 3% of the weld thickness were also made on both sides of the test block weld for shear-wave calibration. Excessive weld spatter on base material adjacent to the dissimilar welds precluded surface contact with the shear-wave transducer, allowing only normal-wave inspection at 2.25 MHz. Conversely, the stainless-steel welds could be tested only by the 45° shear wave because the as-welded surface was too rough to allow intimate transducer contact in the normal mode. Further, the test was limited to directing sound from the pipe side only because the steep slope of the outside surface of the nozzle allowed only a limited scanning distance from that direction.

**Reactor-Vessel Flange.** Both liquid-penetrant and ultrasonic tests were used. The former test was

performed at two locations on the cladding surfaces between the stud holes and on the inner diameter using the same techniques as for the head. No indications were found. The ultrasonic test was made by the normal-wave method from two directions to detect discontinuities between stud holes and in the circumferential weld of the flange to the vessel. A Sperry Model UM-721 instrument was used with a ¾-in.-diameter lithium sulfate transducer and a frequency of 2.25 MHz.

The top surface of the reactor closure flange was found to be rusty and required buffing prior to testing. Because it was desirable to limit the radiation exposure time of the personnel involved, only a portion of this surface was examined by ultrasonics. This surface was scanned in 15 areas between the stud holes. Calibration consisted in using a back-reflection technique. No indications were found.

The second normal-wave search was made from the top of the flange. Calibration was made by using a 5/32-in. flat-bottom hole drilled ¾-in. deep into the end of a bar of similar material 36 in. long—the test distance to the circumferential weld of the flange to the vessel. No indications were found.

**Inlet and Outlet Pipe Welds.** The inlet pipes (downcomers) and outlet pipes (risers) are stainless steel with two welds to the reactor vessel—a bimetallic shop weld and a stainless-steel field weld. Liquid-penetrant examination of both welds in all pipes revealed no indications. The welds between the stainless-steel pipe and the safe end could not be tested by the normal ultrasonic beam because of surface roughness in the as-welded condition. Shear-wave inspection was limited to testing from the stainless-steel-pipe side of the weld owing to interference of the insulation ring on the safe-end side. No indications were found. The dissimilar welds could not be ultrasonically tested because of the coarse-grain condition of the welds.

Higgins[14] describes the General Electric portion of the examination which concentrated on piping welds, control-rod-drive stub-tube welds, and a control rod.

During 1966, several circumferentially oriented defects were found in base metal adjacent to welds in 6-in., schedule 80, seamless, 304 stainless-steel valve-bypass lines in the secondary steam-generator loops and were being monitored on a periodic basis. These two defect indications showed no sign of propagation in the 12 months after detection. In the 1967 outage, two new defect indications were discovered in 6-in. valve-bypass lines. These defects were also circumferentially oriented in base metal adjacent to the weld. They were ground out and weld-repaired. No other defect indications were found during this phase of the inspection. However, during a hydrostatic test of the primary system prior to plant startup, two leaks were observed in the 6-in., schedule 80, stainless-steel equalizer line. This line was not included in the scope of the original planned inspections and was not examined ultrasonically prior to this hydrostatic test. This line contains 16 welds in a total length of approximately 115 ft. Ultrasonic examination of these welds revealed five indications of defects in addition to the two leaking cracks. Again, these were circumferential defects located in base metal adjacent to welds. The entire line was removed and replaced.

With this discovery the inspection program was expanded to include examination of all the accessible remaining pipe welds in 304 stainless-steel pipe 2 to 10 in. in diameter, plus approximately 25% of the circumferential welds in the larger (up to 22 in. in diameter) stainless-steel plate pipe. One defect was detected in the 4-in. reactor-vessel cleanup and drain-supply system located in the vicinity of the equalizer line. This was also a circumferentially oriented indication in base metal adjacent to a weld, and it was removed from the system. Three very small normal-weld-metal defects were also detected on the one 3-in. line. No other defect indications were observed.

In all cases the welds were inspected for both longitudinally and circumferentially oriented flaws by scanning with a ¼- by ¼-in. Automation Industries type SMZ 45° angle-beam, 2.25-MHz transducer in conjunction with a glycerin couplant and a portable ultrasonic tester. Calibration standards were sections of 4- and 6-in., schedule 80, 304 stainless-steel pipe with longitudinally and circumferentially oriented machined notches of depths 10 to 50% of the wall thickness. These standards were used to set the portable ultrasonic tester prior to the start of each sequence of inspections and at periodic intervals thereafter. Surface preparation consisted in removing the insulation, brushing the surface, and applying glycerin as the couplant between the pipe and the transducer.

For record purposes, there is no relation between the *in-service* pipe cracking at Dresden 1 and the *preservice* cracking experienced at Oyster Creek 1, Tarapur, and Nine Mile Point. The latter is related to fabrication and installation heat-treating history of furnace-sensitized 304 stainless steel. The former is a service-related phenomenon peculiar to 7 years of Dresden 1 operation and to a small-pipe welding

process used only in that plant, namely, block welding using a high heat input and a low number of passes.

The stainless-steel stub tubes are welded to the control-rod-drive housings in the 9-in.-thick lower head using a special "J"-groove geometry. The complex shape and lack of weld accessibility required the use of a special ultrasonic-probe assembly. Figure 9 shows the probe in inspection position. The probe head consists of three spring-loaded contact transducers spaced 120° apart: a 45° shear-wave sensor, a 70° shear-wave sensor, and a longitudinal-wave sensor. The transducers were calibrated by use of a mockup housing tube with machined defects of known size. The probe-head-support tube performs three functions in addition to supporting the head:

- It contains the transducer leads.
- It contains the transducer water-couplant lines.
- It is indexed on the surface to permit exact location of the transducers inside the control-rod housing.

The J-groove welds were located with the longitudinal transducer and then inspected for circumferential cracking with 45 and 70° angle-beam transducers over 360° of the circumference. A total of 11 housing welds were inspected by this technique. No indications of defects were detected.

A single boron carbide rod was inspected for longitudinal cracking with a focused-beam immersion transducer attached to a long rod. The rod was placed in V-blocks located on a table immersed in the Dresden 1 fuel pool. The transducer was focused approximately 2 in. from the rod and moved axially while the rod was slowly rotated. No evidence of cracking was found. The literature does not indicate what calibration was performed.

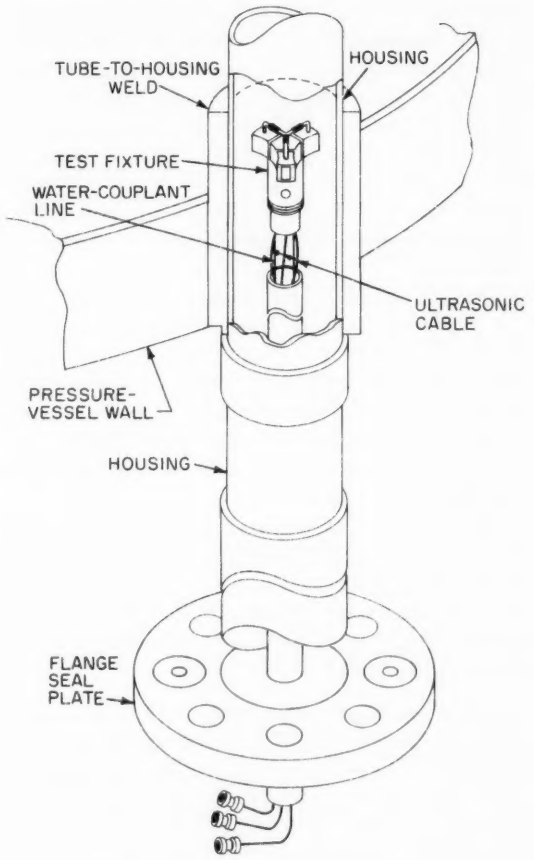


**Fig. 9 Special ultrasonic probe for Dresden 1 weld examination of stub tube to control-rod-drive housing.**

### Elk River

Wylie, Norris, and Whiting[17] and Hawkesworth, Lautzenheiser, and Reinhart[10] describe one of the earliest comprehensive programs in in-service inspection as designed by Southwest Research Institute (SWRI) for the Elk River reactor, a 22-MW(e) BWR owned by the AEC and operated by the Rural Cooperative Power Association, Elk River, Minn. Indeed, SWRI pioneered many of the inspection techniques it uses today during examination of this reactor in 1968. Wylie[9,17] describes a basic "surveillance system," which includes an irradiation-effects surveillance program, a temperature and corrosion monitoring program, and an inspection program. Only the last will be reviewed here as being appropriate to the particular subject under review.

The technical specifications for operation of the Elk River reactor require that certain inspections be carried out approximately every full-power year of operation. These requirements include borescopic examination of the weld-overlay cladding of 8-, 10-, and 16-in. nozzles; direct examination of the overlay cladding at the vessel-shell flange, the top-head flange, and the 12-in. nozzles; and ultrasonic and liquid-penetrant examinations of inlet and outlet primary piping and welds at points of high operating stresses. Until they were replaced by a new design, the boron stainless-steel control rods were to be given a borescopic inspection.

Inspections conducted after the first full-power year of operation revealed cracking in the shell-flange

overlay cladding, which did not penetrate to the base steel. Also, 2 of 13 control rods were cracked in the blade section and 11 had cracked in the retrieval pin gussets. The rods were subsequently replaced with a new design eliminating the boron stainless steel. No evidence of cracking was found in the primary-piping weldments selected for inspection.

Inspections conducted after the second full-power year of operation did not reveal any change in the cracking observed in the shell-flange overlay, nor any cracking at other inspection points in the vessel and primary piping. An ultrasonic inspection of the shell-flange area provided added assurance that the cladding cracks did not penetrate the base steel.

Approximately one-half year after the second inspection, accumulation of water in the lower reactor-vessel cavity indicated a primary-system leak. Locating the leak proved to be a difficult task of 9 months' duration. The leak, in a 1½-in.-diameter stainless-steel upper-liquid-level connection, was a result of intergranular cracking of furnace-sensitized material. Additional inspections of accessible critical areas were carried out to determine if the problem was peculiar to a failed piece or was of a more general nature. Ultrasonic inspections located indications near the ID surfaces of the two 10-in.-nozzle safe ends, the 1½-in. emergency-coolant connection, one primary piping weld that had passed earlier inspections, and one evaporator inlet transition piece. One of the indications in the 10-in. nozzle was confirmed radiographically to be a crack, but confirmation could not be obtained in the other 10-in. nozzle, the piping weld, or the evaporator transition piece. Since the emergency-coolant connection could be removed, it was evaluated metallographically, and the presence of ID cracks was confirmed. Reference 9 includes some detail on the methods and equipment used.

Reference 18 is a typical documentation of the scope and results of inspections on various plant components. In this particular report, both radiographic and ultrasonic tests are described. The sensitivity of ultrasonics is noted, but the problem of locating defects in all orientations remains.

### La Crosse

A series of SWRI reports[19-21] describes programs proposed and carried out to inspect the 50-MW(e) La Crosse BWR and associated piping and equipment of the Dairyland Power Cooperative, Genoa, Wis. Hendricks[19] describes a proposed remote television system for visually inspecting the reactor and supporting systems where conditions precluded direct viewing. Howell and Lautzenheiser[20] describe the design and installation of circular ring tracks and companion electric cars to permit ultrasonic inspection of pipe welds in the recirculation header. This system, the Butt-Weld Inspection Device, is documented in operational use by Whiting.[21] The techniques, standards, and instrumentation used in preservice inspection of selected portions of the primary pressure-boundary components are given in considerable detail.

### Indian Point 1

Consolidated Edison Company's 265-MW(e) PWR, Indian Point No. 1, at Buchanan, N. Y., started commercial operation in 1963. Prestele and Makepeace[22] outline an in-service inspection program that includes: visual and dye-penetrant inspections of the pressurizer internal cladding and the reactor-vessel cladding just below the main closure flange, closed-circuit television inspection of the core shroud, borescope inspection of the reactor-vessel cladding below the lower grid assembly, and magnetic-particle examination and ultrasonic inspection of the reactor-vessel-head studs and selected reactor-coolant-pump flange bolts. The authors describe the methods used to detect tube failures in a steam generator, the means of repair and testing, and the investigation of causes.

Recently a thermal sleeve in a makeup line to one loop failed, causing several pieces of separated metal to lodge in the steam-generator inlet. Subsequent ultrasonic, liquid-penetrant, and metallurgical examinations have confirmed the existence of cracking on the inside diameter of the primary pipe near the makeup-line nozzle. References 23 and 24 describe the problem and the actions and inspections being taken to determine its extent and confirm correction. Complete inspection and repair had not been completed at the time this article was written.

### N.S. *Savannah*

Tillou, Parker, and Thorpe[25] describe the inspections of various portions of the primary system during the first refueling of the ship at Galveston in 1968. The ship had been in service since 1961. The program required the unique application of various well-established inspection methods to evaluate three specific areas:

1. Welded pipe joints.
2. Reactor-vessel internals and structural components.
3. Fuel elements and control rods removed from the core.

Detailed written procedures were prepared for each inspection. The procedures and equipment were checked out using mockups of the components and areas to be inspected. Personnel training included repetitive dry runs using the procedures, equipment, and mockups. All radiation-protection procedures were strictly simulated to prepare adequately for the actual inspections.

The large butt-welded pipe joints were ultrasonically inspected by the angle-beam method, using Lucite transducer shoes contoured to the outside radius of each pipe joint. The smaller diameter joints were dye-penetrant inspected by the visual dye method. There was no indication of cracks or other damage in any joint that was inspected. The reactor-vessel closure studs, nuts, and washers were given a thorough visual inspection using a five-power hand lens. Suspicious areas were further evaluated using dye penetrant. The internal structure of the 48 studs was evaluated ultrasonically using the longitudinal pulse–echo method. The accessible portions of the stainless-steel cladding on the inside of the reactor-vessel head were inspected using water-washable fluorescent dye penetrant applied with a long-handled spray gun. The ultraviolet light was mounted on an articulated wand, and a 90°, 4-in.-diameter nonmagnifying periscope provided visual sighting. There was no indication of cladding damage.

The reactor internals were inspected using a periscope equipped with a 160° panoramic lens. Lighting was provided by a miniaturized quartz iodide light. There was no indication of failure or damage. Similarly, the accessible sections of the inside surface of the reactor-vessel wall were visually inspected using the high-density light. The area was found to be clean and sound.

Highly radioactive fuel elements and control rods (stored underwater), removed from the reactor vessel, were visually inspected using binoculars, a floating-glass viewing box, and a 90° 4-in. magnifying periscope. Dimensions were checked using remotely operated micrometer and special feeler gauges. Illumination was provided by submersible floodlights. No degradation was observed.

### Connecticut Yankee

In a recent report to the AEC,[26] the Connecticut Yankee Atomic Power Company describes an inspection by SWRI of the stainless-steel safe ends in the pressurizer, steam generators, and reactor pressure vessel of the 575-MW(e) Connecticut Yankee reactor at Haddam Neck, Conn. Dye-penetrant inspection and ultrasonic techniques were used to ensure that there was no evidence of stress-corrosion cracking in furnace-sensitized material. Ultrasonic scanning was performed in both the axial and circumferential directions with 45 and 60° shear angles.

Complete inspection was hampered by geometric and metallurgical conditions and by weld spatter and arc strikes. No indications were found of a stress-corrosion cracking condition. Dye-penetrant examination of the weld between a steam-generator nozzle and safe end did show axial indications lying across the Inconel weld bead and across the junction of the stainless-steel-to-Inconel weldment. These were traced back to a bracket weld made during construction and subsequently removed. It is a good indication of the state of the art when mention is made that, because of geometry and metallurgical conditions, adequate inspection could not be made in all directions. Much more developmental work is obviously needed before true volumetric inspection can be assured.

### Brunswick 1 and 2

Units 1 and 2 of Carolina Power & Light Company's 821-MW(e) BWR in Brunswick County, N. C., will go into commercial operation in 1974 and 1976, respectively. The two-unit station is noted because its proposed in-service inspection program[27] is a good example of the type of planning each power-reactor operator will have to do in the course of obtaining construction and operating permits. The reference uses Section XI of the ASME Code as the program guide. As for many reactors of this time period, their design did not consider Section XI requirements and thus some improvisation will be necessary owing to limited accessibility. Major emphasis will be on ultrasonic methods, supplemented by visual and magnetic-particle and liquid-penetrant methods. Because of the particular dry-well design, there is limited access around the reactor vessel. Conceptual design of freestanding reflective insulation indicates 6 to 7 in. is the available annulus for equipment. Carolina Power & Light has commissioned SWRI to design an ultrasonic inspection device to fit in this space.

### Foreign Reactors

References 28 and 29 summarize the worldwide status of inspection methods for reactor pressure vessels as of 1966. The reader will find little on in-service inspection because the emphasis is on manufacturing and preservice inspections.

Hawkesworth, Lautzenheiser, and Reinhart[10] describe systems developed for reactors in Sweden, Argentina, and Italy. In general, the SWRI methods reviewed earlier were adapted mechanically to the particular designs of each reactor.

## DESIGNING FOR IN-SERVICE INSPECTION

A consistent theme throughout the literature on operating experience with in-service inspection is: design the system initially to allow in-service inspections to be made quickly and completely. Brandt[30] describes these provisions as needed in BWRs. The article, although written 2½ years before the issuance of Section XI of the ASME Code, contains many of the basic code requirements. Included are comprehensive tables of in-service inspection locations; methods and frequency of inspections; and suggested BWR-design provisions, which, in general, include:

—Adequate clearances.

—Allowance for erection of temporary working platforms.

—Direct-access crawl routes.

—Removable insulation and a place to store it.

Anderson, Forrer, and Hubs[31] suggest these design and manufacturing and construction changes for economical, effective inspections:

—As aids in visual and volumetric inspections, all surfaces should be properly prepared by the manufacturer, e.g., smooth weld surfaces, peelable painted coatings, and smooth contours.

—Since the major areas for inspection are welds, a minimum number of welds results in a more economical, faster inspection. The use of forgings in lieu of plate is a step in the right direction..

—Designs that minimize abrupt changes in cross section should be provided.

—There is no specific standard for the format of records. However, records must be legible and interpretable over the operating period. Standards such as quality of paper and format legend need to be developed for these records systems.

—Power for ultrasonic and magnetic-particle equipment, water for decontamination and liquid-penetrant examination, and air for grinders and repair equipment should be provided permanently in the general vicinity of each of the major areas selected for inspection. These services are not normally provided in the inspection area.

—It is prudent to provide a reasonable storage- and inspection-area stand that allows proper maneuvering since the closure head will normally be off the vessel during inspections.

—Insulation is applied to all systems to reduce heat loss. This insulation, however, interferes with the performance of the inspection. Further attention and development are necessary to permit easy removal and alternate placement of this insulation for inspection purposes.

—Bench marks or other means of identification should be used to aid reproducibility of test results. Precise as-built information is a must. Base-line inspection data should be obtained, preferably using the same techniques and equipment that will later be used. Drawings should show accurately the location of welds, supports, and other attachments.

—There should be proper recognition of and preparation for expected radiation levels.

In a recent paper, Foster[32] of Gilbert Associates gives a comprehensive outline of all plant design and construction factors to be considered in preparing for full compliance with Section XI of the *ASME Boiler and Pressure Vessel Code.* Although the paper was prepared to describe the architect–engineer's role in in-service inspection, it does give for the first time a comprehensive set of guidelines to meet the code requirements at minimum cost from design through operation. Proper planning from the earliest time of contract award is stressed—and with the total involvement of reactor and plant design, quality assurance, construction, inspection, maintenance, and operations.

The reader must recognize that, at the present time, the true implications and requirements for full compliance with Section XI of the Code are not known. That can only come as actual experience is gained over the next several years. In fact, the AEC is currently requiring applicants to define their in-service inspections programs only for the initial 5 years of operation, to permit the acquisition of experience and to take advantage of developments in this field. This requirement will make possible a more practical inspection program to be defined during the remaining service years.

## INSPECTION-PERFORMANCE CONSIDERATIONS

Holyoak, Forrer, and Higgins,[14] on the basis of the Dresden 1 experience, describe in vivid terms what is involved in performing in-service inspections. Adequate

prior planning is the key. When the factor of radiation, with its requirement for protective clothing and masks and time-restricted dose limitations, enters the picture, the schedule that is logical for a shop inspection must be revised considerably. In meeting schedules, it is customary traditionally to work longer hours if the schedule is falling behind. However, to meet a schedule in nondestructive-testing work in radiation areas with a limited number of trained people, it becomes more important to revamp techniques to obtain the same results more quickly rather than to work longer hours.

The participants should thoroughly review the areas to be inspected, prepare all the equipment needs, and make necessary work assignments so that the inspection can be performed in the most precise, economical manner. The participants include:

—Mechanics to provide access for inspections and to secure the inspection areas.

—Health physicists to monitor personnel radiation exposure.

—Inspection personnel.

Since the first two groups should be plant personnel, it is important to schedule their activities so that they will be available to provide support to the inspection teams in an efficient manner without interfering with refueling operations.

Anderson, Forrer, and Hubs[31] point out that, owing to lack of experience, the most difficult inspection to plan is the first one. They suggest that proper planning is not possible without a radiation map of the system to determine requirements for inspection manpower, test equipment, area shielding, decontamination, protective clothing, and health-physics support.

Selig, Kosky, and Hedden[13] estimate the manpower requirements and personnel exposures for a complete 10-year inspection program as follows: a 15-man nondestructive-testing crew would be required for the examinations performed in parallel with annual refuelings; 25 men would be required for approximately 20 days of around-the-clock work at the 10th-year examinations when the vessel internals are removed. Over the 10-year period, the authors estimate 13,500 man-hours, including health-physics and supervisory personnel. It is speculated that the total exposure would be about 70 man-rems with an average exposure of 0.35 to 0.47 rem/man/inspection, which is well below the maximum allowable limit of 3 rems per quarter.

The authors note that in 1968 Duke Power Company estimated a 47-day extension in downtime over 10 years for nonautomated inspection systems. Comparison is then made with their estimate of 11 to 13 days for automated systems. With outage costs for an 1100-MW(e) plant of $50,000 to $100,000/day, the incentive for automated systems is strong. An equally important reason that was not mentioned is that automation is a necessity, at least in some areas, because of local radiation fields.

## SUMMARY

On Nov. 24, 1969, the AEC announced its intention to amend its reactor-licensing regulations to include sections on codes and standards as they relate to systems and components important to safety. In a parallel action, the American Society of Mechanical Engineers issued a new *Boiler and Pressure Vessel Code,* Section XI, In-Service Inspection of Nuclear Reactor Coolant Systems. The conjunction of both these events brought to the attention of industry with startling rapidity the need for in-service inspection and proper planning for their use.

Some in-service inspections have been conducted on U. S. and foreign reactors; however, considerable development and field experience are needed to demonstrate adequate compliance with the code and with minimum impact on plant operating schedules and cost.

## ACKNOWLEDGMENTS

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# Current BWR Fuel Design and Experience

By H. E. Williamson* and Dana C. Ditmore†

**Abstract:** *An up-to-date review of experience and development data on boiling-water-reactor (BWR) Zircaloy-clad $UO_2$ fuel is presented. The area of technology demonstrated by irradiations of Zircaloy-clad $UO_2$ fuel rods and pins in reactors, loops, and capsules goes beyond the combination of fuel-rod power and exposure that will be experienced in modern BWRs not only for normal continuous operation but also for anticipated power transients. The recorded successful irradiations demonstrate that not only safe but reliable fuel can be designed for modern BWR conditions. A value of 1% plastic strain of Zircaloy cladding is defined as the limit below which fuel damage due to overstraining is not expected to occur during any single power transient. The linear heat-generation rate required to cause this amount of cladding strain is ~28 kW/ft in fresh fuel. The analytical and experimental bases employed in the justification of this limit are reviewed.*

## INTRODUCTION

Water-cooled-reactor fuel experience and testing[1,2] over the past several years have provided the technology for continued evolution of fuel designs of higher thermal performance and exposure capability. This increase in the fuel-performance capability in conjunction with other technological advances has been applied to advantage in modern boiling-water-reactor (BWR) plants by increases in the design core power density up to 40 to 50 kW/liter. The corresponding performance conditions for the fuel are a peak thermal power for normal continuous operation of 17.5 to 18.5 kW/ft [linear heat-generation rate (LHGR)] and a design peak exposure of 45,000 MWd/metric ton.[1]

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The coolant environment has remained essentially unchanged since the first experience was obtained; i.e., the coolant pressure is ~1000 psig, and the coolant temperature does not exceed ~288°C, even in the hottest channel at overpower. The only significant change in environment for the BWR has been a marked reduction in the amount of corrosion-product impurities introduced in the feedwater as a result of improvement in feedwater system design.

The fuel-rod diameters and cladding wall thicknesses have remained essentially unchanged between the first production fuel rods placed in service in the Dresden 1 reactor in 1960 and the higher performance fuel going into plants today. The significant changes have been conversion from segmented to full-length fuel rods and increased void volume in the rods for $UO_2$ expansion and fission-gas release associated with longer design exposures and higher thermal performance.

Experience gained in irradiations of Zircaloy-clad $UO_2$ fuel rods and pins in water-cooled reactors, loops, and capsules over the past 10 years has demonstrated successful operation beyond the combination of fuel-rod power and exposure that will be experienced in modern BWRs not only for normal continuous operation but also for anticipated power transients. Many more data at these combinations of fuel-rod linear power and burnup will become available in the future; however, the recorded successful irradiations demonstrate that not only safe but reliable fuel can be designed for modern BWR conditions.

The large volume of experience with Zircaloy-clad $UO_2$ production fuel operated in BWRs, supplemented by test irradiations that extend up to and beyond the proposed operating conditions,[1] forms an important base for the current fuel designs. First, this experience

demonstrates the capability to manufacture production quantities of fuel with appropriate skill and quality control. Second, and technologically more important, all this experience has been accumulated on fuel rods very similar in design to current fuel and operated in essentially the same coolant environment (pressure, temperature, and chemistry); thus the applicability of the experience is ensured.

The increase in fuel-performance capability in the modern BWR is exemplified by the proposed operating limits on fuel-rod power for the Dresden 2 and Browns Ferry classes of reactors, respectively. The proposed operating limits for fuel-rod power are 17.5 kW/ft (LHGR in fresh fuel) for the Dresden 2 class of reactors (e.g., Dresden 2 and 3, Quad Cities 1 and 2, and Millstone) and 18.5 kW/ft (LHGR in fresh fuel) for the Browns Ferry class of reactors (e.g., Browns Ferry 1, 2, and 3; Peach Bottom 2 and 3; and Vermont Yankee). During the worst anticipated power transients, resulting from equipment malfunction or operator error, the overpower LHGR can reach peak local values as high as 20 to 22 kW/ft in fresh fuel (18.0 kW/ft in highly depleted fuel). However, the fuel and core design criteria have been defined to assure sufficient operating margins exist to accommodate such anticipated transient power increases without experiencing damage that might reduce fuel lifetime.

This article summarizes the current BWR fuel design, highlights the areas of major design conservatism incorporated to assure safe and reliable service, and reviews the large volume of experience on Zircaloy-clad $UO_2$ production and developmental fuel that forms the base for the current BWR fuel design. The information in the article is primarily based on the General Electric BWR fuel design and design bases because of the availability of pertinent documentation.* Essentially all the experience obtained to date in power operating BWRs has been with fuel designed and manufactured by General Electric Company.

## THE BWR FUEL ASSEMBLY

A typical BWR fuel assembly operates for about 5 years at an average power level of 4 MW(t). During this time the fuel will reach an average exposure level of 25,000 MWd/ton. A single assembly may produce as much as 7 MW of thermal power when the BWR core is operating at peak rating.

Figure 1 is a schematic of a typical BWR fuel assembly. The salient features of this design include the fuel-assembly channel, upper and lower tie plates, upper handle, fuel rods, and interim fuel-rod spacers. Table 1 summarizes the basic data on fuel dimensions available in the open literature.[1,3]

Forty-nine fuel rods are loaded into a square 7 by 7 array called a fuel-rod bundle. The fuel rods contain uranium dioxide enriched in $^{235}U$ in the form of

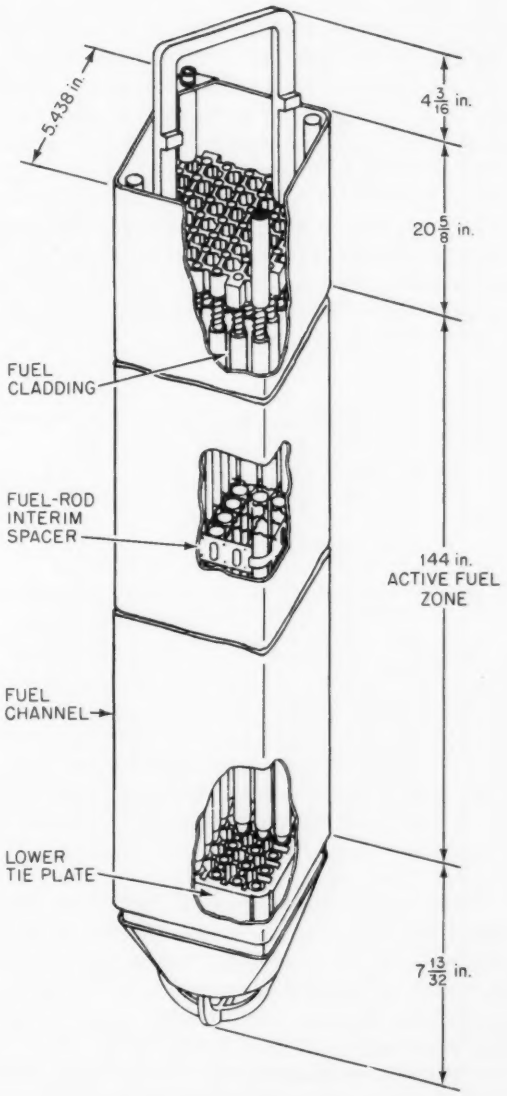


Fig. 1 Fuel-assembly schematic.

*USAEC Public Document Room, 1717 H Street, Washington, D. C. 20545.

high-density pellets. The rods are spaced and supported in the square 7 by 7 array by upper and lower tie plates of stainless steel and interim fuel-rod spacers of Zircaloy-4. The lower tie plate has a nosepiece that supports the fuel assembly in the reactor. The upper tie plate has a handle for transferring the bundle from one location to another.

Eight fuel rods are used to provide the structural tie between the upper and lower tie plates. Each one of these rods threads into the lower tie plate and extends through the upper tie plate, where nuts are threaded and tightened down. The nuts maintain the upper tie plate in its proper position. It is also spaced from the fuel rods by coil springs on the shank of each end plug. All other rods are captured between the upper and lower tie plates, each rod having upper and lower end plugs that fit into holes machined in the respective tie plates. For insertion into and removal from the core, as well as for other handling maneuvers, the fuel assembly is picked up by the upper-tie-plate handle—its weight supported by the tie rods. Because of its design, the fuel assembly can be disassembled readily to expedite repair on the site in the event of damage during normal handling operations.

**Table 1 Fuel Data**

| | |
|---|---|
| Fuel-assembly data | |
| Overall length, in. | 175.87 |
| Nominal active fuel length, in. | 144 |
| Fuel-rod pitch, in. | 0.738 |
| Number of fuel rods | 49 |
| Space between fuel rods, in. | 0.175 |
| Fuel-channel wall thickness, in. | 0.080 |
| Fuel-bundle heat-transfer area, $ft^2$ | 86.52 |
| Fuel-rod data | |
| Outside diameter, in. | 0.563 |
| Cladding thickness, in. | 0.032 |
| Pellet outside diameter, in. | 0.487 |
| Fission-gas plenum length, in. | 16.00 |
| Pellet immersion density, $g/cm^3$ | 10.42 |

Each fuel-rod bundle is enclosed by a surrounding square Zircaloy-4 channel that fits over the lower tie plate and extends to the upper tie plate. The channel provides a confined path for coolant flow. The fuel-rod bundle and channel, together with tie plates and spacers, comprise a single fuel assembly. The fuel assemblies are loaded in the BWR core to approximate a right-circular cylinder. The BWR core is made up of a repetition of modules (Fig. 2), the number of which is

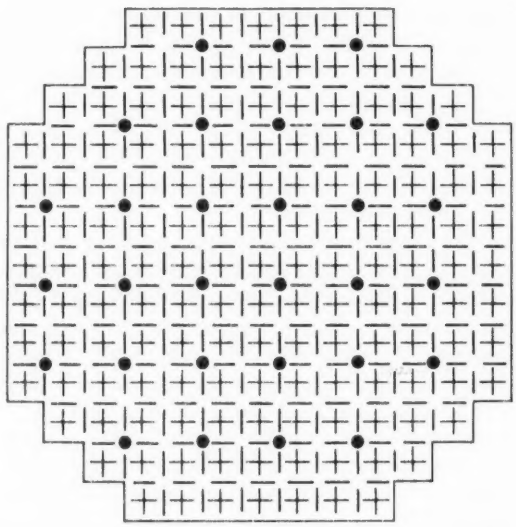

**Fig. 2 Typical core arrangement. +, control blades; –, curtains; •, in-core monitor location.**

established by the heat-producing capability required for a particular power plant. A module is comprised of four fuel assemblies surrounding a control blade (Fig. 3). Coolant flow enters each fuel assembly through the lower-tie-plate nosepiece and flows vertically upward between the fuel rods. Cooling of other core components is maintained by allowing a controlled amount of leakage, or bypass, coolant flow in the space between fuel assemblies. The cruciform control blade is movable, and the reactivity and power distribution within the core are controlled by its axial position. Supplementary reactivity control is needed with fresh fuel and is provided either in the form of boron stainless-steel-alloy curtains positioned between fuel assemblies or by gadolinia, mixed with $UO_2$, in selected fuel rods within the fuel assemblies. The neutron-absorbing boron in the curtains burns out during operation as the fresh-fuel reactivity decreases with depletion. Once the burnable poison $^{10}B$ is consumed, the curtains are removed from the core. The gadolinia in urania is also a burnable poison. However, when the gadolinia isotopes of high absorption cross section have been consumed, the gadolinia-bearing fuel rods remain in the core to produce additional thermal power.

When the control blade is withdrawn, the water in the space between the fuel assemblies causes some neutron-flux peaking due to increased neutron modera-

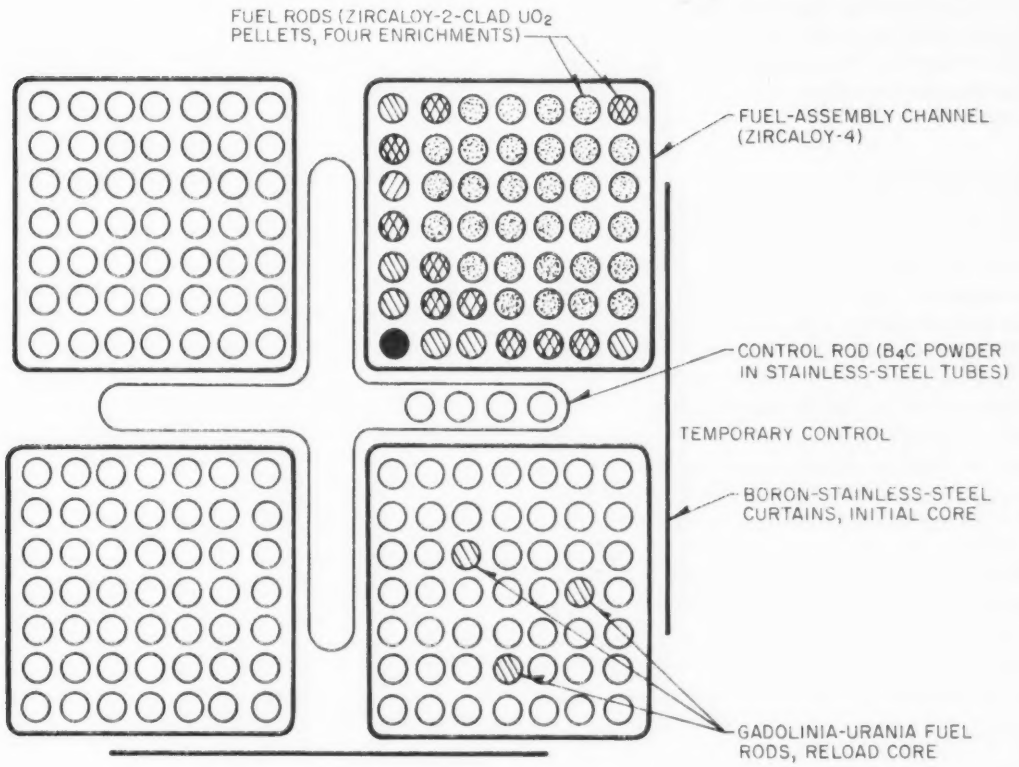
FUEL RODS (ZIRCALOY-2-CLAD $UO_2$ PELLETS, FOUR ENRICHMENTS)
FUEL-ASSEMBLY CHANNEL (ZIRCALOY-4)
CONTROL ROD ($B_4C$ POWDER IN STAINLESS-STEEL TUBES)
TEMPORARY CONTROL
BORON-STAINLESS-STEEL CURTAINS, INITIAL CORE
GADOLINIA-URANIA FUEL RODS, RELOAD CORE

Fig. 3 Typical fuel module.

tion in the area of the control-blade water gap. The $^{235}U$ enrichment of the fuel rods is varied across a fuel assembly in such a fashion as to even out the power effects resulting from this condition. The lower enrichment fuel rods are placed in the corner adjacent to the control blade, with those of higher enrichment in the center of the assembly.

Fission heat generated within the fuel rods transfers to the coolant (water) flowing vertically upward between the fuel rods. The coolant flow becomes a saturated liquid anywhere from one-eighth to one-third of the distance between upper and lower tie plates. Beyond this point bulk nucleate boiling of the coolant occurs, and the transferred fission heat converts a portion of the saturated liquid coolant to saturated vapor (steam). After exiting the fuel assembly through the upper tie plate, the saturated steam–water mixture is separated into its phases, i.e., saturated vapor and liquid at about 1000 psig. The saturated vapor is available to drive a steam turbine; the saturated liquid is combined with returning feedwater and recirculated to the bottom of the BWR core.

In a BWR core there are significant gradients in power from the core center to the core extremities in both the radial and axial direction and also a significant gradient in power across individual fuel assemblies. As a consequence of these gradients, not quite 1% of the fuel elements in a core should, at a given time, experience the design peak operating condition of 17.5 kW/ft (LHGR) for the Dresden 2 class of reactors and 18.5 kW/ft (LHGR) for the Browns Ferry 1 class of reactors. Further, the proposed operating limits do not represent the anticipated long-term, continuous operating conditions. It is expected that the peak operating limits of 17.5 and 18.5 kW/ft for the Dresden 2 and Browns Ferry 1 classes of reactors, respectively, will be experienced by individual fuel rods at different times during each operating cycle. However, the majority of the fuel elements in a reactor core is not expected to achieve this condition, and the long-term, average

power condition for the pellet that attains peak exposure (~45,000 MWd/metric ton) will be in the range of only 7 to 12 kW/ft (LHGR) over the entire life of the peak fuel element.

## THE BWR FUEL ROD

### Design Description

The basic element of the modern BWR fuel design is a Zircaloy-2-clad, cylindrical, $UO_2$-pellet fuel rod. Each fuel rod consists of high-density (95% of theoretical) ~½-in.-diameter uranium dioxide fuel pellets, enriched with $^{235}U$, stacked in a 144-in. column, and enclosed within a 0.563-in.-diameter Zircaloy-2 cladding tube that is evacuated, backfilled with helium, and sealed by welding Zircaloy end plugs in each end (Fig. 4). Uranium is used in the dioxide form because this compound is thermally and chemically stable in the reactor environment. It reacts only at a very slow rate with boiling water at reactor operating conditions. The fuel-rod cladding thickness, 32 mils, is adequate to be "freestanding," i.e., capable of withstanding the external reactor coolant pressure without collapsing onto the enclosed fuel pellets. Adequate free volume is provided within each fuel rod to accommodate the helium backfill plus all gaseous fission products released from the fuel pellets during operation without excessive internal pressure. Dished fuel pellets are employed in the highest power, highest exposure fuel rods to accommodate long-term irradiation-induced swelling of the $UO_2$. All other fuel rods employ a standard right-circular cylindrical pellet with no dish. The dish is a small depression in each end of a dished pellet (Fig. 4). The fuel-pellet-to-cladding initial diametral gap is 12 mils. A summary of pertinent fuel-rod data is given in Table 1.

### Basis for Design

The basic objective for BWR fuel design is to provide assembly of fissionable material of high structural integrity, capable of efficiently transferring generated fission heat to the circulating coolant water while containing fission products and fuel material over the intended lifetime. The fuel design details and operating limitations specifically provide a large operating margin to the postulated fuel-rod-damage condition to assure that the basic design objective will be met. Further assurance is gained through application of design models that give a pessimistic representation of cladding loadings and the use of lower limit values to

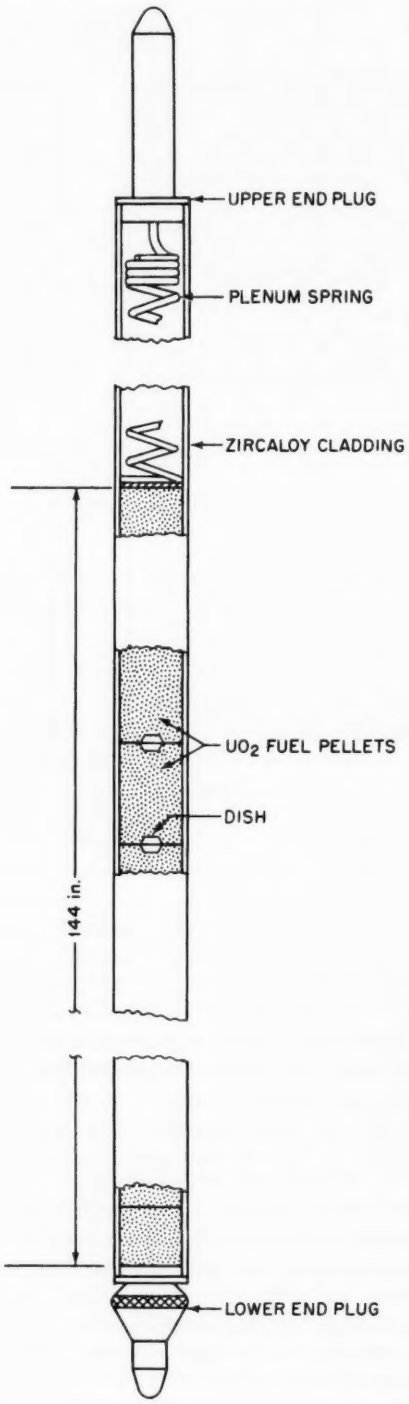


Fig. 4 Fuel-rod schematic.

represent cladding-strain capability in the definition of fuel damage.

The design features and performance limitations are established through consistent application of stringent design criteria that reflect the intent of the basic design objective. The design criteria require that:

1. The fuel-rod-cladding external design pressure shall be equal to or greater than that which would occur during either anticipated normal or transient operation.

2. The internal design pressure of the fuel-rod cladding shall exceed that resulting from initial fill gas, impurity gases, and released fission-product gases throughout the intended fuel lifetime.

3. The cladding strain caused by relative expansion between fuel and cladding shall be a small fraction of that which will cause perforation.

In the application of these fundamental fuel-design criteria, all material stresses are maintained within specified design limits for normal anticipated modes of operation and abnormal operational transients up to incipient center melting of the fuel material. Table 2 is a summary of the BWR stress-intensity limits.

**Table 2 Stress-Intensity Limits**

| Categories | Yield strength (Sy) | Ultimate tensile strength (Su) |
|---|---|---|
| General primary-membrane stress intensity | $\frac{2}{3}$Sy | $\frac{1}{2}$Su |
| Local primary-membrane stress intensity | Sy | $\frac{1}{2}$ to $\frac{3}{4}$Su |
| Primary-membrane-plus-bending stress intensity | Sy | $\frac{1}{2}$ to $\frac{3}{4}$Su |
| Primary-plus-secondary stress intensity | 2Sy | 1 to $\frac{3}{2}$Su |

In satisfying these design limits, the fuel is capable of short-term operation at conditions in excess of those required to cause incipient fuel melting without experiencing fuel damage. However, exceeding the design limit by operation of the current fuel design beyond incipient fuel melting could reduce fuel lifetime.

Fuel damage is defined as perforation of the cladding that permits release of fission products. Two primary mechanisms may cause fuel damage during reactor transients:

1. Severe local overheating of the fuel cladding caused by inadequate cooling.

2. Fracture of the fuel cladding due to excessive strain resulting from relative pellet/cladding expansion.

Fuel damage due to local overheating of the fuel cladding is defined as the onset of the transition from nucleate to film boiling, i.e., minimum critical-heat-flux ratio (MCHFR) equal to unity. However, on the basis of critical-heat-flux (CHF) data, fuel damage is not actually expected to occur until well into the film-boiling regime. The basis for the definition of this damage limit is well documented in existing BWR safety-analysis reports.[3]

A value of 1% plastic strain of Zircaloy cladding is defined as the limit below which fuel damage due to overstraining is not expected to occur during a transient. Available data indicate the threshold for damage in irradiated Zircaloy cladding is in excess of this value. The LHGR required to cause this amount of cladding strain is approximately 28 kW/ft in fresh fuel of the current design and approximately 22 kW/ft at a local exposure of 40,000 MWd/metric ton.

In comparison, the LHGR required to cause incipient fuel melting is 21.5 kW/ft, whereas the anticipated peak LHGR during normal modes of steady-state operation in the current generation of BWRs is only 18.5 kW/ft.

### Design Considerations

The designing of nuclear fuel, in general, is a complex process that must predict the interaction between predominant thermal/mechanical mechanisms and then incorporate design features which will allow for accommodation of resulting cladding loading. Generally, the most predominant mechanisms affecting cladding loading can be related to two basic considerations: (1) release of gaseous fission products from the fuel pellet and (2) relative axial and radial expansion between fuel and cladding.

The release of gaseous fission products from the fuel serves to increase the fuel-rod internal pressure. The relative expansion between fuel and cladding also affects internal gas pressure (axial expansion) and can result in strain of the cladding due to pellet-to-cladding interaction (relative radial expansion) in the maximum-duty fuel rods.

These two basic considerations are influenced either directly or indirectly by the performance of the fuel during operation. The power generation in a fuel assembly decreases over its lifetime in relation to fresher and more reactive fuel inserted in later refuelings (*A* in Fig. 5). The power generation within a fuel assembly varies from rod to rod and varies over its life (*B* in Fig. 5). As previously explained, multiple enrichments are used in the assembly to minimize

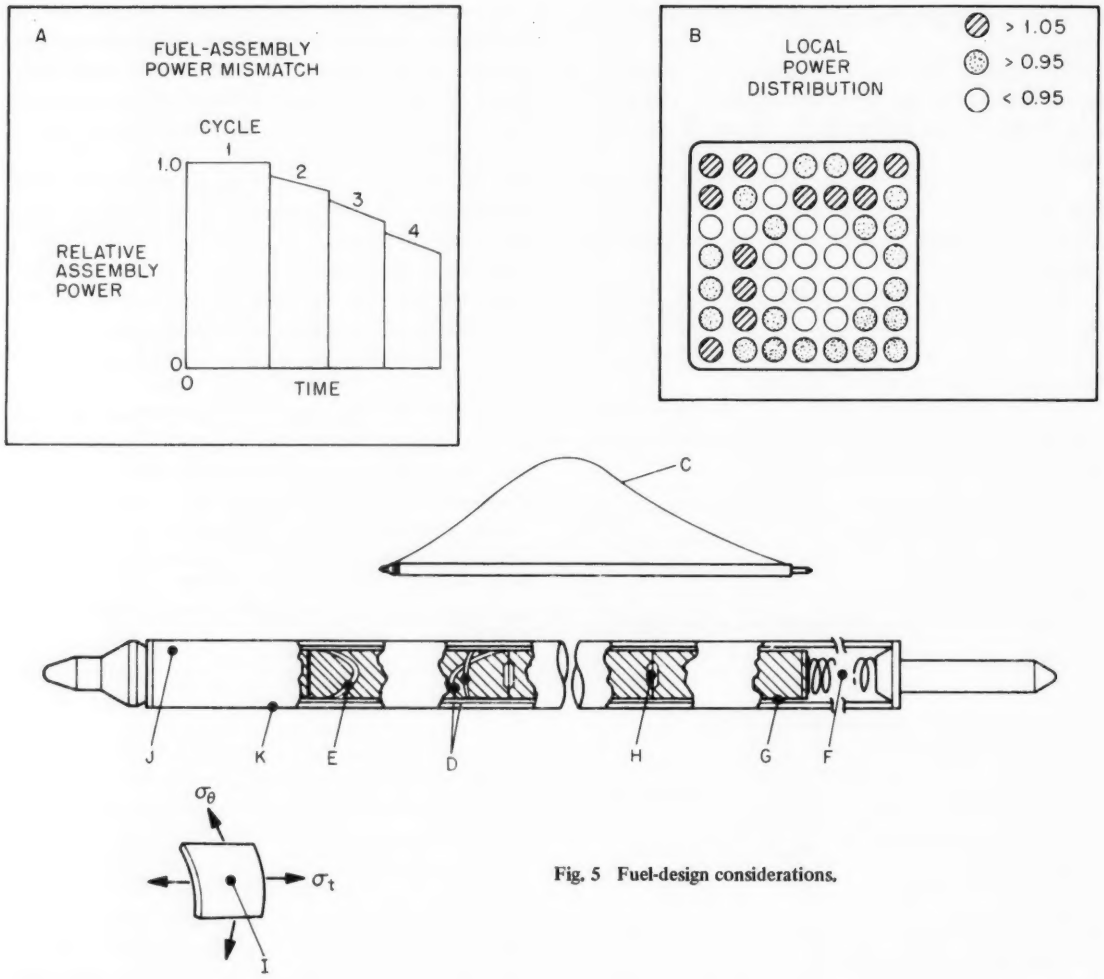


Fig. 5 Fuel-design considerations.

power-generation differences. The fuel-rod internal power generation varies over the length of the rod (*C* in Fig. 5) and radially across a fuel pellet (*D* in Fig. 5).

The $UO_2$ temperature distribution (*E* in Fig. 5) will vary with these power distributions, and the temperature will, in turn, affect fission-gas release and relative fuel/cladding expansions. The plenum at the top of the fuel rod (*F* in Fig. 5) takes care of fission-gas release and differential axial expansion between fuel ($UO_2$) and cladding. A gap between the pellet and cladding (*G* in Fig. 5) accommodates the relative expansion between fuel and cladding in a radial direction. The $UO_2$ porosity and a dish in the pellet (*H* in Fig. 5) provide local open volume for internal irradiation swelling of the $UO_2$. A dish is required because the allowable amount of porosity is limited by the degrading effect of increased porosity on the $UO_2$ thermal conductivity. As indicated previously, dished pellets are provided only in those rods where they are needed by service requirements. Economics dictates that dishing be kept to a minimum since blank space cannot generate energy.

Both ductility and strength of the cladding (*I* and *J* in Fig. 5) must be considered in design, and these properties are a function of irradiation and temperature. The buildup of films on the outside surface of the cladding (*K* in Fig. 5) can be a significant factor in the determination of fuel-cladding temperature. This effect, coupled with the reduction of heat generation over the lifetime of the fuel, makes cladding temperature a strong function of service.

### Design Conservatism

The designs and operating limits of current BWR Zircaloy-clad $UO_2$-pellet fuel have been established using design models that conservatively predict the effects of the predominant thermal/mechanical mechanisms resulting in cladding loadings. Reliance on the combination of conservatism in the design models for prediction of fuel performance and in the definition of cladding mechanical capability assures that adequate margin is maintained between operating limits and the transient cladding-damage limit.

**Fission-Gas Release.** Fission gas is generated within the fuel at a rate of $1.35 \times 10^{-3}$ gram moles/MWd. Fission-gas release is calculated according to the following model: 4% release for $T < 1650°C$, 100% release for $T \geq 1650°C$. That is, 4% of the generated gas is released for fuel temperatures below 1650°C, and 100% of the generated gas is released for fuel temperatures above 1650°C. This model predicts a greater amount of fission-gas release than is shown by actual fission-gas-release data. Figure 6 shows the calculated percentage of fission-gas release for an entire fuel rod vs. the average fuel temperature compared to actual data from selected prototype fuel rods containing $UO_2$ pellets that had been irradiated under the AEC Fuel Cycle Program and other AEC programs.[4,5] Fission-gas-release data from these tests are of particular interest since the design of the prototype fuel and the irradiation environment to which this fuel was subjected were reasonably close simulations of those encountered in current BWR designs. Figure 6 does not represent all previously reported data but does represent the data considered most applicable to the fuel design and the operating conditions expected in currently proposed BWRs. Most data fall in the range of 2 to 30% release. For a maximum centerline temperature of $\leq 2800°C$ ($UO_2$ melting) for the fuel rod, the release[5] was no greater than 30%. By comparison the BWR fuel-design model would predict maximum release rates as high as 60% for similar conditions with a maximum centerline temperature of 2800°C. Reported data[5,6] for conditions of gross centermelt indicate a maximum release of only 60%.

Within the exposure range covered by the data of Fig. 6, there is no discernible effect of burnup on the release rate. Included in Fig. 6 are data from the SA-1 fuel assembly with rod average exposures estimated to

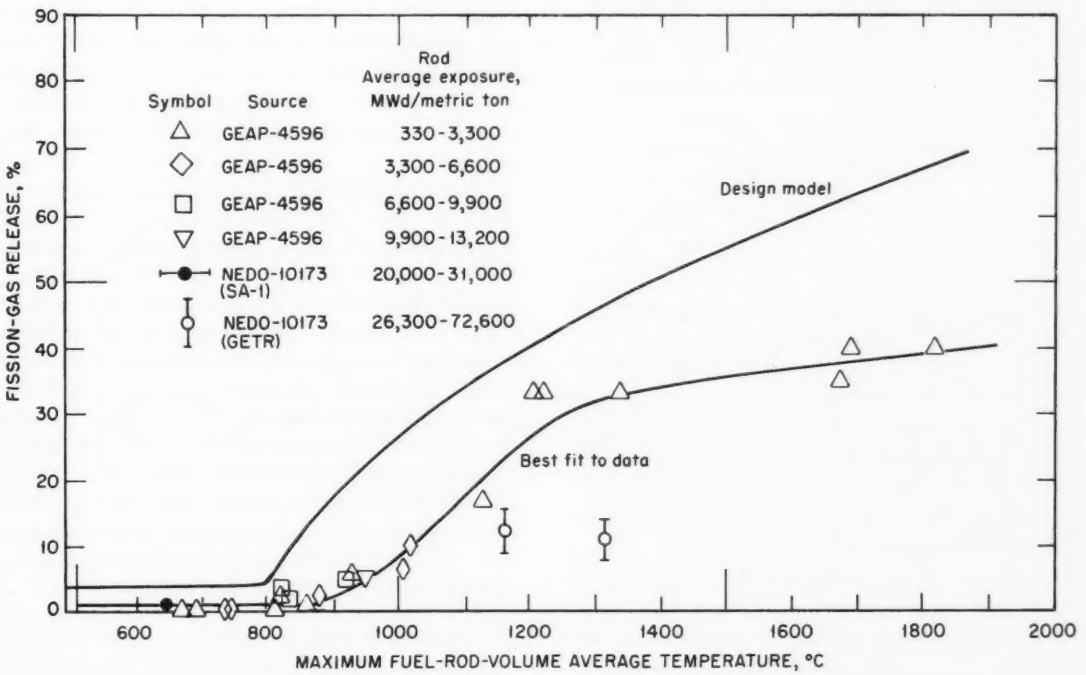


**Fig. 6 Comparison of predicted/observed fission-gas release (from Ref. 1).**

be in the range of 20,000 to 31,000 MWd/metric ton and recent data from high-power/high-exposure General Electric Test Reactor (GETR) capsules with rod average exposures to 72,600 MWd/metric ton.[1] The band on the SA-1 data point indicates the estimated range on average fuel temperature. All data for SA-1 have less than 1% release fraction.[4] The lack of apparent burnup dependence in the in-pile fission-gas-release data has been attributed to the large temperature-dependent releases masking the small amount of low-temperature, time-dependent gas release.[5]

**Fuel/Cladding Relative Expansion.** Free volume is provided in the fuel pellets and between pellets and cladding to accommodate the anticipated relative fuel and cladding expansions in a diametral direction without experiencing excessive cladding strain and in an axial direction without experiencing excessive cladding stress due to plenum-volume reduction. Both fuel ($UO_2$)-pellet expansion due to irradiation swelling and thermal expansion and cladding thermal expansion are considered in the design calculation.

The BWR design model for determination of relative fuel/cladding expansion employs material properties,[3] such as rates of thermal expansion as a function of temperature for fuel material ($UO_2$) and cladding (Zircaloy), which are consistent with available data for these materials. The fuel-material ($UO_2$) irradiation-swelling model calculates fuel-irradiation swelling as a combination of that due to the accumulation of both solid and gaseous fission products. Initially the fuel is assumed to swell at a rate of 0.16% $\Delta V/V$ per $10^{20}$ fissions/cm$^3$ consistent with the Bettis data on flat-plate $UO_2$ fuel.[7] When the capacity of the internal pellet porosity and dish volume is reached owing to a combination of both solid and gaseous fission-product swelling, the rate of total external pellet swelling increases sharply to values ranging from ~0.4 to as high as ~1.0% $\Delta V/V$ per $10^{20}$ fissions/cm$^3$, the exact value depending on fuel temperature, burnup, fission rate, etc. The basis for the gaseous fission-product swelling model is the work of Greenwood and Speight,[8] incorporating the Van der Waals gas equation of state and the effect of external hydrostatic pressure. The model is very similar to the model employed in pressurized-water-reactor (PWR) design.[9]

The BWR design model for determination of relative fuel/cladding expansion has been compared with available experimental data from irradiations of Zircaloy-clad $UO_2$-pellet fuel covering a wide range of power and exposure. These comparisons show that the design model predicts, for Zircaloy-clad $UO_2$-pellet-fuel designs similar to those of current BWRs, relative fuel/cladding expansions which are, in general, greater than are inferred from postirradiation measurements. The degree of conservatism associated with the overall fuel/cladding expansion model, incorporating both thermal expansion and irradiation swelling, is illustrated by the plots of Figs. 7 and 8. These two figures show the ratio of predicted pellet-diameter change to the pellet-diameter change inferred from measured cladding strains (P/O), with calculated adjustment for cladding thermal expansion and elastic springback. Figure 7 represents the ratio of predicted to "inferred" pellet expansion (P/O) vs. exposure, and Fig. 8 shows the same parameter as a function of the LHGR of the test fuel rods. The bands on the data points for Bett 79-21 and 79-25 indicate the range of P/O for a number of measurements taken at different axial locations on the fuel rods at each exposure.

One of the most significant sources of data on postirradiation fuel measurements is the Bettis–LSBR/LWBR Development Program.[10-14] In particular, WAPD-TM-629 (Ref. 11) provides many detailed data taken before, during, and after irradiation examinations of rods Bett 79-21 and 79-25. The LHGR for these test rods is as high as that predicted for end-of-life in BWR fuel, and the exposures are even higher. In addition to demonstrating less fuel expansion than predicted by the BWR design model (see Figs. 7 and 8), the data in Ref. 11 also show the effectiveness of the use of dished ends on $UO_2$ pellets to reduce diametral growth due to fission-product swelling.

The fuel assembly with the highest exposure to date in a commercial power BWR is the SA-1 fuel assembly, which has reached an average assembly exposure of ~30,000 MWd/metric ton in Dresden 1. Postirradiation data are being reported in a series of USAEC research and development reports.[4] Application of the design fuel/cladding expansion model to the SA-1 performance predicted that some pellet-to-cladding interactions would have occurred, but, from the postirradiation measurements, it was not possible to conclude that any cladding strain due to pellet expansion had occurred. In fact, the measured postirradiation dimensional variations in the cladding diameter were of the same order of magnitude as initial tubing tolerances and oxide thickness on the fuel.

**Cladding Mechanical Capability.** Demonstration of the conservatism in the definition of 1% plastic strain of irradiated Zircaloy as a transient damage limit is available from mechanical-property tests. Tests were

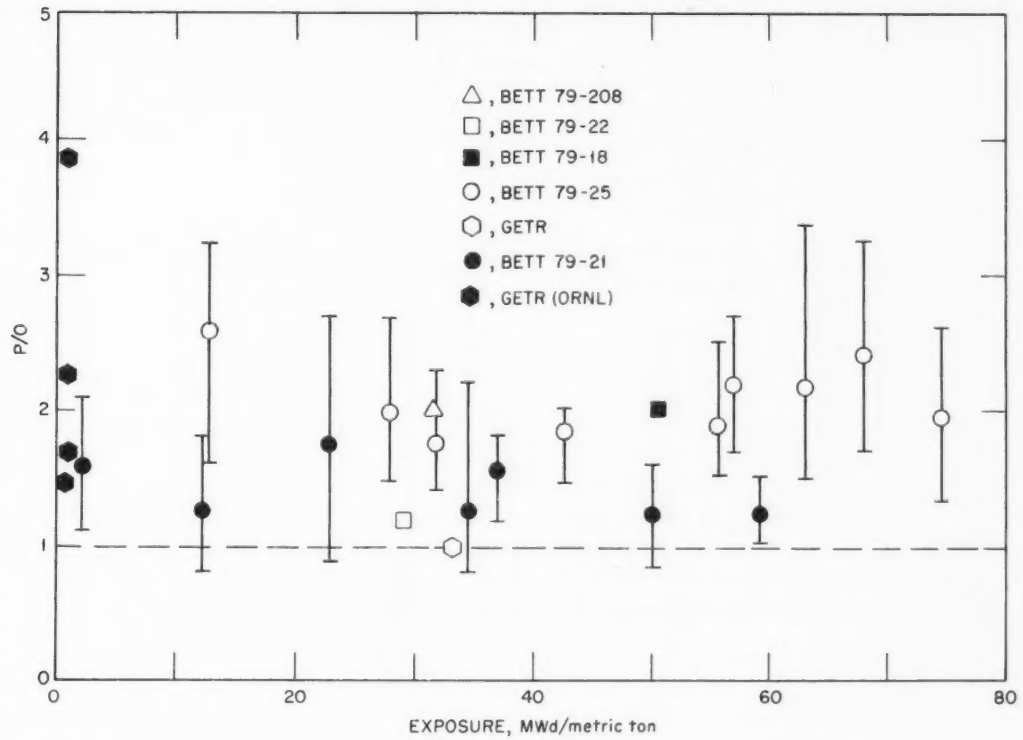


**Fig. 7 Predicted/observed (P/O) pellet-diameter change vs. exposure (from Ref. 1).**

performed on samples taken from actual irradiated Zircaloy-2 fuel cladding. Included were cladding samples from Dresden Nuclear Plant (DNP) Type I, AEC Fuel Cycle Program, SA-1, and DNP Type IIIF fuel. Values for total plastic elongation in all instances were 1% or greater.[1]

The integrated neutron fluxes of the samples tested were less than the expected end-of-life integrated flux of about $10^{22}$ neutrons/cm$^2$, but the test properties are considered representative of end-of-life owing to saturation of irradiation effects. Figure 9 shows the effects of fast-flux exposure on elongation and includes data from the AEC Fuel Cycle Program and SA-1. This figure shows that saturation occurs at 0.3 to $0.4 \times 10^{21}$ neutrons/cm$^2$ (>1 MeV), equivalent to about 3000 to 4000 MWd/ton fuel exposure.

Dresden Type I fuel cladding was tested in the form of full tube segments in which the fuel had been removed and in the form of miniature tensile coupons machined from tube segments. Testing in all cases was in the longitudinal tube direction. The fuel rods from which the cladding samples were removed had exposures ranging from 13,800 to 15,800 MWd/metric ton. All cladding samples were from nonfailed fuel rods. Results are shown in Table 3.

Detailed examinations were performed on Zircaloy-clad fuel rods from the AEC Fuel Cycle Program; these rods had been irradiated in the Vallecitos Boiling Water Reactor (VBWR) to a calculated average exposure as high as 10,000 MWd/metric ton. Tensile specimens having integrated fast fluxes ranging from about 0.56 to $1.4 \times 10^{21}$ neutrons/cm$^2$ were machined from fuel-cladding segments. Specimens were all oriented such that testing was in the longitudinal tube direction. Results[15] are shown in Table 4.

Also examined[4] in detail were Zircaloy-2-clad SA-1 fuel rods that had been operated to a cladding fluence of $0.4 \times 10^{22}$ neutrons/cm$^2$ (>1 MeV) in Dresden 1. Tensile tests on longitudinal-coupon cladding specimens gave total elongation values ranging from 1.0 to 3.0%. The results of these tests are shown in Table 5.

Elevated-temperature burst tests were performed on cladding-tube segments from irradiated DNP Type IIIF reload fuel. Tube segments were cut from

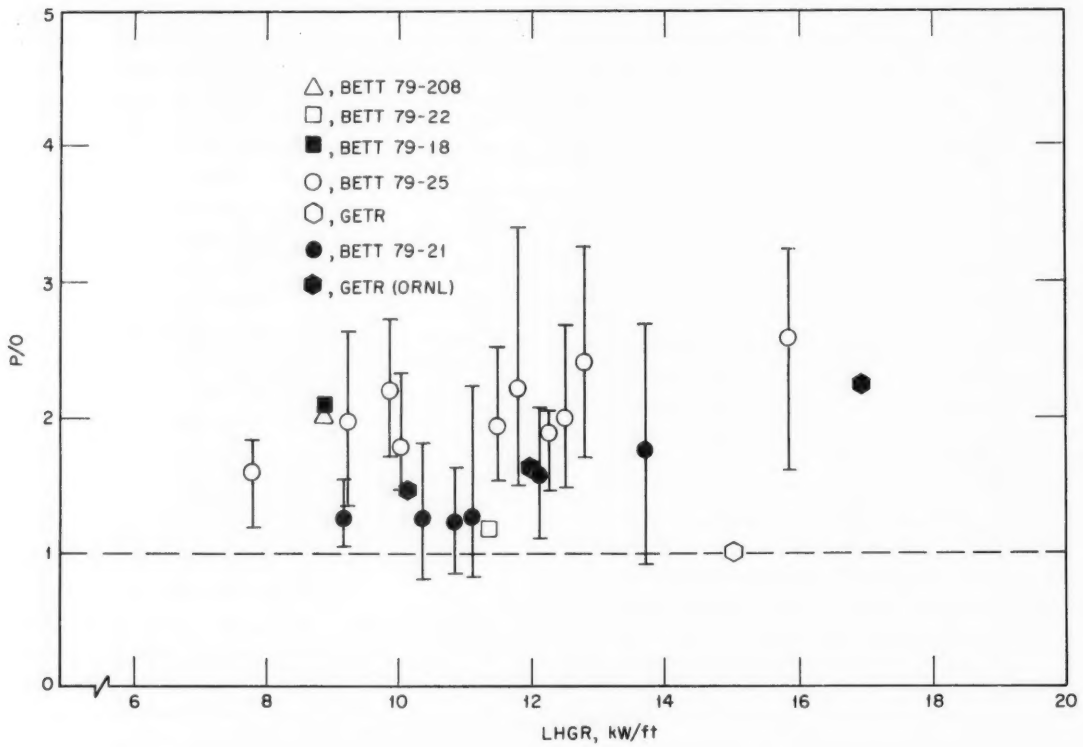


**Fig. 8 Predicted observed (P/O) pellet-diameter change vs. linear heat-generation rate (LHGR) (from Ref. 1).**

irradiated fuel rods, and the $UO_2$ fuel was removed. The ends of each tube segment were welded closed with end plugs, one of which had a special fitting for pressurization of the tube. Burst hoop stress and plastic circumferential elongation were measured. Results are shown in Table 6.

Of all the tests mentioned above, the burst results on DNP Type IIIF fuel cladding are considered to be the most representative data on strain to fracture for current tubing.

The burst tests on DNP Type IIIF fuel cladding provide a measure of tube properties, under biaxial loading, in the transverse or circumferential tube direction and thereby eliminate the uncertainties caused by the anisotropic nature of zirconium.

The DNP Type IIIF fuel cladding was produced according to the specifications required for current BWRs. The minimum acceptable elongation is twice that of the minimum elongation in the DNP Type I fuel cladding (16% vs. 8%) primarily because of improvements in the production process. The DNP Type I and AEC Fuel Cycle Program claddings were produced from extrusions by several reductions through the tube-reduction process followed by drawing operations of 10 to 20% reduction in area (draw-to-size). In the case of the AEC Fuel Cycle Program, the tubing received a final stress-relief anneal. Current production practice is tube reduction in three or four steps to the final dimensions (rock-to-size), with a last reduction in area of 50 to 70% followed by a stress-relief anneal. Cladding produced by this rock-to-size process has greater ductility and finer grain size than cladding made by the draw-to-size process. In addition, the DNP Type I cladding was tested using eddy-current techniques prior to development of superior ultrasonic methods. Subsequent ultrasonic testing of DNP Type I cladding samples revealed multiple small defects which had gone undetected and which were expected to have a detrimental effect on available ductility.

The results of all the above-mentioned tests on irradiated BWR cladding indicate total plastic elonga-

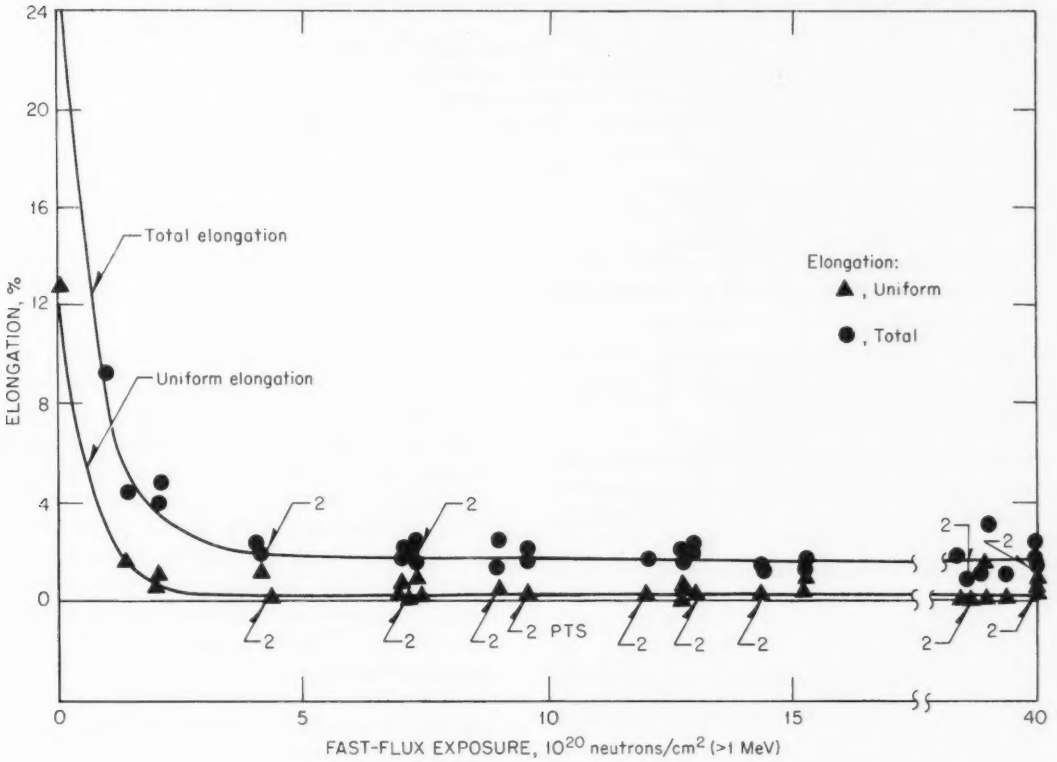


**Fig. 9 A 345° tensile elongation of irradiated Zircaloy-2 and Zircaloy-4 cladding samples from AEC Fuel Cycle Program and SA-1 fuel (from Ref. 1).**

**Table 3 Tensile Tests on Zircaloy-2 Cladding of Dresden Type I Fuel***

| Temperature, °F | Yield strength (0.2% offset), ksi | Ultimate tensile strength, ksi | Uniform elongation, % | Total elongation, % |
|---|---|---|---|---|
| | Full Tube Specimens [1.8 – 2.7 x $10^{21}$ neutrons/cm$^2$ (> 1 MeV)] | | | |
| RT | 99.1 | 112.1 | 4.8 | 6.8 |
| RT | 87.8 | 109.9 | 4.0 | 5.4 |
| 650 | 60.3 | 74.8 | 2.0 | 3.2 |
| 650 | 56.9 | 74.5 | 2.9 | 4.6 |
| 650 | 50.9 | 71.9 | 2.3 | 2.8 |
| 750 | 48.8 | 68.2 | 2.6 | 3.9 |
| | Coupon Specimens [1.7 – 2.2 X $10^{21}$ neutrons/cm$^2$ (>1 MeV)] | | | |
| 650† | 68.9† | 71.7† | 1.6† | 3.98† |
| 650‡ | 6.12–76.4‡ | 66.6–80.4‡ | 1.2–1.9‡ | 2.7–7.6‡ |
| 800 | 58.6 | 61.0 | 0.9 | 4.6 |
| 800 | 59.6 | 61.9 | 0.9 | 4.3 |

*From Ref. 1. †Average of 10 tests. ‡Range.

Table 4 Tensile Tests on Zircaloy-Clad Fuel from the AEC Fuel Cycle Program*

[0.56 to 1.4 x $10^{21}$ neutrons/cm$^2$ (>1 MeV)]

| Temperature, °F | Yield strength (0.2% offset), ksi | Ultimate tensile strength, ksi | Uniform elongation, % | Total elongation, % |
|---|---|---|---|---|
| RT† | 80.6† | 76.4† | 0.98† | 3.17† |
| RT‡ | 58.7–104‡ | 86.2–107.4‡ | 0.38–2.8‡ | 1.4–5.3‡ |
| 650† | 56.7† | 57.7† | 0.26† | 1.67† |
| 650‡ | 51.4–66.0‡ | 52.0–66.5‡ | 0.13–0.80‡ | 1.0–2.3‡ |

*From Ref. 1. †Average of 21 tests. ‡Range.

tion greater than the 1% value employed in the definition of the strain-damage limit for Zircaloy cladding in power transients.

**Damage Limit.** Up to this point the discussion has centered on BWR fuel-design models and cladding-strain capabilities. Use of the design models permits calculation of the LHGR required to cause fuel damage (1% plastic strain of the cladding) during an overpower transient. The damage-limit LHGRs have been calculated[1] and are given in Table 7 as a function of fuel exposure. The damage limit corresponding to 1% plastic strain in the fuel cladding is calculated to be ~28 kW/ft in fresh fuel (0 MWd/metric ton exposure) of the current design.

For comparison, the anticipated peak steady-state and transient conditions are shown for both the Dresden 2 and the BF-1 classes of reactors. Both the anticipated operating conditions and the damage limits decrease monotonically with burnup. The power-generating capability of the fuel decreases owing to depletion of fissionable isotopes as burnup increases. Partially depleted fuel assemblies will operate at a decreasing LHGR as fresh reload fuel is inserted in each cycle. Owing to long-term irradiation effects, including $UO_2$ swelling, the damage limit corresponding to 1% plastic strain also decreases. These effects are illustrated in Table 7. The table also shows that there is a wide margin to the transient fuel-damage condition maintained throughout life.

A number of development fuel rods and capsules prototypical of current designs have been successfully operated at LHGRs in excess of 28 kW/ft without cladding damage. Perhaps the most significant to date is the recent experience with one fuel pin in Capsule A of Table 10 (which will appear later). This fuel pin was operated for an extended period (about 2 weeks) at a peak LHGR of 29.4 kW/ft, after having achieved an exposure of 27,000 MWd/metric ton, without experiencing failure.

The capsule continued to operate successfully until the fuel pin had achieved an exposure of approximately 34,000 MWd/metric ton, at which time a failure did occur. Visual examination of the failure showed a cladding crack. The failure appeared to be due to the

Table 5 Tensile Tests at 650°F on Zircaloy-2 Cladding for SA-1 Fuel*

| Neutrons/cm$^2$ (>1 MeV) | Elongation, % | |
|---|---|---|
| | Uniform | Total |
| 3.8 x $10^{21}$ | 0.7 | 1.2 |
| 3.8 x $10^{21}$ | 0.4 | 1.4 |
| 3.8 x $10^{21}$ | 0.3 | 1.2 |
| 3.8 x $10^{21}$ | 1.1† | 2.2† |
| 3.4 x $10^{21}$ | 0.3 | 1.1 |
| 3.4 x $10^{21}$ | 1.7† | 3.0† |
| 3.2 x $10^{21}$ | 0.2 | 1.0 |
| 3.6 x $10^{21}$ | 0.3 | 1.1 |
| 3.2 x $10^{21}$ | 0.3 | 1.2 |
| 3.1 x $10^{21}$ | 0.2 | 1.7 |

*From Ref. 1.
†Accuracy of these values is in doubt. Specimens may have slipped in grips.

Table 6 Burst Tests on Cladding of DNP Type IIIF Fuel*

[Strain Rate 2.4 x $10^{-2}$ in./(in.)(hr)]

| Temperature °F | Neutrons/cm$^2$ (>1 MeV) | CFE,† % | Hoop stress, psi |
|---|---|---|---|
| 650 | 0.8 x $10^{21}$ | 6.22 | 92,700 |
| 650 | 0.8 x $10^{21}$ | 7.35 | 94,900 |
| 900 | $10^{21}$ | 13.56 | 69,300 |

*From Ref. 1. †Plastic circumferential elongation.

Table 7 Fuel Linear Heat-Generation Rates*

| Exposure, MWd/metric ton | Peak steady state | | Anticipated peak transient | | Cladding-strain damage limit |
|---|---|---|---|---|---|
| | D-2 | BF-1 | D-2 | BF-1 | |
| 0 | 17.5 | 18.5 | 19.9 | 21.5 | 28 |
| 22,000 | 16 | 17 | 18.2 | 19.7 | 28 |
| 44,000 | 14 | 15 | 15.9 | 17.4 | 22 |

*From Ref. 1.

repeated cyclic operation with significant degrees of central melting, for which the fuel pin was not designed. The prior sustained operation in excess of 28 kW/ft at 27,000 MWd/metric ton did not result in a detectable failure. Two of the fuel rods[6] irradiated under the High Performance $UO_2$ Program (Table 9, which will appear later) showed 1% plastic strain of the Zircaloy cladding to correspond to approximately 28 kW/ft; these fuel rods did not fail. Two other fuel rods were operated to 14,000 MWd/metric ton at peak powers as high as 58 kW/ft (LHGR) without failure; these two fuel rods, which had $UO_2$ fuel pellets with a small (0.1-in. diameter) center hole to accommodate the $UO_2$ phase-change volume increase on melting, successfully operated at LHGRs in excess of 28 kW/ft without cladding damage.

## EXPERIENCE

Much experience with Zircaloy-clad $UO_2$-pellet fuel has been obtained over the past 10 to 12 years. Most of this experience was in commercial power BWRs operating at somewhat lower peak LHGRs than are currently required. However, a significant amount of experience at high-thermal-performance conditions has been obtained with smaller quantities of fuel under various development programs in both boiling-water and pressurized-water test reactors throughout the world.

The large volume of production experience, starting with the first load of fuel in Dresden 1 in 1960, has provided feedback on the adequacy of the design for, and the effects of, operation in a commercial power-reactor environment. Production fuel experience has also provided feedback on the incidence and effect of flaws and impurities that occur statistically in large-volume production processes.

The developmental experience has demonstrated the capability of the fuel-rod design to meet the increased performance requirements of the modern BWR system. Prototypical fuel rods and assemblies have been irradiated in both power and test reactors to either high-peak local exposures (~44,000 MWd/metric ton) or to high-peak LHGRs (>28 kW/ft). Pressurized (1000 psi) boiling-water fuel capsules containing short fuel pins have been operated to peak local combinations of exposure and power which go beyond that to be experienced by fuel in the modern systems.

The design models for performance-limiting characteristics are evaluated by comparison with detailed tests and observations on the irradiated fuel rods and pins. This experience has been used in establishing design features and in the analysis of performance characteristics. The following presents an up-to-date review of current and past experience with high-performance BWR Zircaloy-clad $UO_2$-pellet fuel that has formed the basis for the modern BWR fuel designs, fuel operating limits, and fuel-damage limits.

### Experience with Operating Production Fuel[1]

Table 8 is a summary of leading BWR experience with currently operating production Zircaloy-clad $UO_2$-pellet fuel. Overall, more than 12 production fuel types have been designed, manufactured, and successfully operated in eight operating commercial power BWRs. When all production fuel types are considered,* a total of more than 218,000 Zircaloy-2-clad $UO_2$ fuel segments have been operated with assembly average exposures as high as 23,500 MWd/metric ton and peak local exposures in excess of 33,800 MWd/metric ton. Failures due to wall perforation have been detected in ~0.1% of these segments, including failures in fuel that had exceeded its design performance conditions.

The cladding perforations have been investigated in detail, the causes identified, and corrective measures instituted to reduce the probability of future recurrence. The causes of the observed failures can be divided into three basic categories: design limitations, manufacturing-related problems, and external effects of the plant system on the fuel.

Cladding failures due to design limitations have not been observed except on the Dresden 1 initial fuel. The predominant number of cladding perforations have been from manufacturing-related causes and external effects from impurities in the circulating coolant water. The design-related failures have been eliminated by stress-relieving the cladding tubing to prevent bowing and by increasing the space in a fuel rod for $UO_2$ expansion and fission-product gases.

*Only currently operating fuel is given in Table 8.

**Table 8 Summary of Leading Experience on Operating Production Zircaloy-Clad $UO_2$-Pellet Fuel***

| Name | Exposure,† MWd/metric ton (peak pellet/ average per assembly) | Time† in core, years | Max. heat flux,‡ § Btu/(hr)(ft$^2$) | Peak LHGR, kW/ft‡ § | Fuel-rod dia., in. | Cladding thickness, mils | Pellet-to-cladding gap (nominal), mils | Active fuel length, in. | Fission gas plenum (volume per unit fuel volume) | Number of segments or rods |
|---|---|---|---|---|---|---|---|---|---|---|
| Dresden Type I¶ | 33,800/23,500** | 9.25 | 350,000 | 15.2 | 0.567 | 30 | 3 | 106.5 | 0.013 | 144 |
| Dresden Type IIIB | 29,600/15,400 | 4.95 | 360,000 | 15.4 | 0.555 | 35 | 7.5 | 109.0 | 0.040 | 5,868 |
| Dresden Type IIIF | 27,800/15,600 | 3.95 | 360,000 | 15.5 | 0.5625 | 35 | 10 | 108.25 | 0.048 | 3,360 |
| Dresden V | 15,700/ 7,400 | 1.95 | 360,000 | 15.5 | 0.5625 | 35 | 10 | 108.25 | 0.048 | 3,816 |
| Garigliano Type I | 24,700/12,700 | 6.40 | 252,000 | 10.3 | 0.534 | 30 | 5 | 105.7 | 0.031 | 11,502 |
| Garigliano Type II | 8,800/ 5,400 | 1.20 | 320,000 | 14.6 | 0.593 | 37 | 11 | 107 | 0.030 | 3,456 |
| KAHL | 21,400/11,000†† | 7.60 | 296,000 | 13.0 | 0.569 | 33 | 5 | 59.8 | 0.017 | 1,368 |
| Humboldt | | | | | | | | | | |
| Type II | 20,300/11,700 | 4.00 | 325,000 | 12.1 | 0.486 | 33 | 10 | 79.0 | 0.043 | 6,468 |
| Type III | 5,060/ 2,500 | 0.75 | 389,000 | 16.8 | 0.563 | 32 | 11 | 79.0 | 0.062 | 1,872 |
| KRB | 17,100/10,100 | 3.10 | 367,000 | 15.8 | 0.5625 | 35 | 10 | 130.0 | 0.058 | 10,512 |
| JPDR | 8,200/ 3,800 | 5.60 | 205,000 | 13.0 | 0.557 | 30 | 5 | 56.75 | 0.100 | 5,184 |
| Consumers | | | | | | | | | | |
| BRP-B | 29,500/19,500 | 3.70 | 434,000 | 15.0 | 0.449‡‡ | 34 | 8 | 70.0 | 0.048 | 1,694 |
| BRP-E | 13,400/ 6,600 | 1.50 | 410,000 | 17.7 | 0.5625 | 40 | 11 | 70.0 | 0.048 | 2,926 |
| BRP-EG | 5,300/ 3,500 | 0.75 | 410,000 | 17.7 | 0.5625 | 40 | 11 | 70.0 | 0.048 | 1,694 |
| Tarapur | 2,700/ 1,400 | 0.85 | 365,000 | 15.8 | 0.5625 | 35 | 10.5 | 144.0 | 0.059 | 27,832 |
| Current BWR§§ | 45,000/27,500 | 5.0 | 428,000 | 18.5 | 0.563 | 32 | 11 | 144.0 | 0.11 | 18,032 |

*From Ref. 1.
†Dresden and JPDR data of Sept. 5, 1969. All other data, except that for KAHL, are as of Dec. 31, 1969.
‡At rated power.
§ License limit.
¶A total of 100 assemblies (14,400 segments) discharged Jan. 1, 1967, with average exposure of 12,600 MWd/metric ton.
**Lead Type I assembly.
††As of Aug. 31, 1968.
‡‡A total of 109 rods with 0.449-in. diameter, 12 corner rods with 0.344-in. diameter.
¶¶Typical design as opposed to proved performance in preceding entries.

Cladding failures caused by excessive amounts of system corrosion products depositing on the fuel should not occur in modern BWR plants because of improved feedwater treatment and the selection of stainless steel instead of copper-based alloys for heat exchangers in the primary system. Cladding failures have been caused by fretting from small wires that escaped from the steam-separator demisters and were trapped in the fuel-assembly spacers. These will not recur because the use of this type of demister has been discontinued for the modern BWR plants.

Manufacturing-related failures observed have been caused by flaws in the cladding tubing and internal decontamination of fuel rods with hydrogenous impurities. Improved nondestructive testing methods for cladding tubing and special process precautions are expected to virtually eliminate this type of failure in the future. All these cladding failures and their causes are dealt with in more detail in subsequent sections on experience in specific plants.

Peak LHGRs, from ~10 kW/ft to ~16 kW/ft, have been achieved with the production fuel. Individual fuel assemblies have been successfully operated for more than 9 years' in-core residence. In addition, during 1969, 3566 fuel rods were placed in service as reload fuel in the Big Rock Point and Humboldt Bay reactors, and more than 65,000 fuel rods were placed in service or loaded as initial core fuel in three new General Electric reactors: Tsuruga (Japan), Jersey Central, and Niagara Mohawk. These fuel rods are all typical of the current fuel design. Since no appreciable exposure has been attained, they are not listed in Table 8. In comparison, the current designs have the following proposed operating characteristics:

17.5 to 18.5 kW/ft maximum LHGR (operating limits).
~45,000 MWd/metric ton maximum local exposure.
30,000 MWd/metric ton maximum assembly exposure.
4 to 6 years' in-core residence time.

Fuel-rod diameters in the range of 0.425 to 0.567 in. in outside diameter, with cladding wall thicknesses from 30 to 35 mils, and pellet-to-cladding gaps from 3 to 11 mils have been used in production fuel. Active fuel-column lengths have varied from 59.8 to 144.0 in. with fission-gas plenum volume per unit of fuel volume from 0.013 to 0.100. In comparison, the currently proposed designs for the Dresden 2 and Browns Ferry reactor types have the following design features:

Fuel-rod outside diameter, 0.563 in.
Cladding wall thickness, 0.032 in.
Pellet-to-cladding gap, 11 to 12 mils.
Active fuel lengths, 144 in.
Fission-gas plenum volume, per unit of fuel volume, 0.08 to 0.11.

At least four new BWRs (Dresden 2, Millstone, Nuclenor, Monticello) employing fuel typical of this design were expected to go into service during 1970. Therefore more than 100,000 additional BWR Zircaloy-clad $UO_2$-pellet fuel rods would be introduced into service for operation up to 17.5 kW/ft and with exposures typical of those indicated above for the Dresden 2 and Browns Ferry classes of reactors.

**Dresden Nuclear Plant (DNP) 1.** Dresden 1 provides the largest body of operating experience on Zircaloy-clad production fuel. As of Sept. 5, 1969, the lead Dresden 1 Type I (DNP Type I) fuel assembly had been operating in-core for more than 9 years and had accumulated an average exposure of 23,500 MWd/metric ton with an estimated peak segment exposure of 33,800 MWd/metric ton and peak LHGR of 15.2 kW/ft.* At the 1967 refueling of Dresden, 14,400 segments of this fuel were discharged with an average exposure of 12,600 MWd/metric ton.

Overall, the experience with DNP Type I fuel has been very successful. Extensive "sipping"† and visual inspection of indicated "leakers"† have shown failures‡ in only 22 segments out of the total 77,184 Zircaloy Type I segments operated in Dresden 1 since 1960 (<0.1% failures). Ten of these failures occurred during the fourth operating cycle (May 1965 to February 1967).

Five of the failures in the first two fuel cycles were due to accelerated local corrosion. The Type I fuel cladding was not stress relieved. During reactor operation the relaxing of the residual stress permitted the

*The peak LHGRs (power) indicated in the text and Table 8 are the license-allowable values. They do not represent average operation but probable peaks during continuous operating conditions.

†A leaker fuel assembly is identified by sipping, or sampling the water surrounding an individual fuel assembly. If analysis of the sipped water indicates fission products are leaking from one or more fuel rods, the assembly is classified as a leaker. A failed fuel assembly, on the other hand, is a leaker in which the source of fission-product release has been located and identified as a cladding wall perforation.

‡A "failed" fuel assembly is one that contains at least one fuel rod with a perforated cladding wall. When used in this sense, a failed fuel assembly can even be one that has been operated beyond design exposures or power-generation rates.

corner fuel rod to bow and touch the channel wall. Coolant flow was locally restricted, and the cladding temperature increased enough to cause cladding failure due to accelerated local corrosion. Stress relief of the cladding and use of the through-rod instead of the segmented-rod design in subsequent fuel have prevented the recurrence of this type of failure.

The second set of five failures in the first two fuel cycles was due to cracking of the cladding from internal corrosion. A fabrication defect in a fuel rod permits water to penetrate the rod and cause internal corrosion. The corrosion products by their volume can strain the cladding to cracking after extended periods of operation. It is believed that, in these five DNP Type I segments, water entered the fuel rods through stringers in the end plugs. Improved bar-stock quality of the end plugs and discontinuance of the end-plug design susceptible to stringers (female screw connectors) have eliminated this particular type of cladding perforation.

The failures observed in 12 segments after the fourth cycle appear to be similar to earlier ones. These 12 failures occurred predominantly in fuel operating in the central portion of the core at about twice the design exposure. The failures are attributable to features that have been improved and corrected in present designs. The fuel was beyond design exposure; inadequate void space for fuel expansion and fission-gas release for the service conditions experienced has been deduced as the cause of most of these failures. Dresden 1 fuel Types IIIB, IIIF, and V are essentially prototypical of current BWR fuel. As of the September–October 1969 refueling outage, average exposures up to 20,500 MWd/metric ton with peak LHGRs of 15.4 kW/ft had been attained on Type III fuel assemblies. As of Sept. 5, 1969, average exposures of 10,900 MWd/metric ton, peak exposures in excess of 15,500 MWd/metric ton, and peak LHGRs of 15.5 kW/ft had been attained on Type V fuel assemblies. Operation of the IIIB, IIIF, and V assemblies is continuing.

Out of nearly 400 of these IIIB, IIIF, and V Zircaloy-clad $UO_2$-pellet reload fuel assemblies operated in Dresden, only 5 had been identified as leakers prior to 1969. Underwater inspection of 4 of these 5 revealed no fuel-rod failures. The fifth assembly was found to have cracked at the bottom end-plug weld in one of the rods. It was concluded that manufacturing defects caused these leaks. The 5 leakers in 400 assemblies could result from imperfections in 5 out of 13,000 fuel rods (<0.1% defects).

A few manufacturing defects should not be unexpected in any production-quantity fabrication process. On the basis of operating experience with intentionally defected rods in the VBWR, the fission-product activity released from a manufacturing-type imperfection is so low as to be of virtually no consequence. However, extended exposure in the defected condition may lead to further cladding perforations by splitting.

During the September–October 1969 refueling outage in Dresden 1 (end of cycle 6), 100% of the core fuel assemblies were sipped. There were 3 Type IIIB, 19 Type IIIF, and 7 Type V production-type $UO_2$-pellet fuel assemblies identified as leakers. Preliminary visual examination at the site suggested that these failures were similar in nature to recent KRB nonfret failures (see discussion of 1969 refueling outage at KRB). A more detailed examination and an investigation are planned.

**Garigliano (SENN).** There are a number of fuel types currently operating in the Garigliano reactor. As of Dec. 31, 1969, the Garigliano Type I fuel had accumulated the highest exposures; 11,502 segments were operating with an average exposure of 12,700 MWd/metric ton and a peak local exposure of 24,700 MWd/metric ton. The peak operating LHGR for this fuel was 10.3 kW/ft. The SENN or Garigliano plant has operated since 1963 with an excellent record. One fuel assembly with a failed fuel rod was detected and removed during the summer 1968 shutdown. The failure was due to cladding fretting caused by a broken in-core chamber and was thus unrelated to the fuel performance. The Garigliano Type II reload fuel was loaded in 1968. As of Dec. 31, 1969, there were 3456 fuel rods operating with an average exposure of ~5400 MWd/metric ton and a peak local exposure of 8800 MWd/metric ton. The peak LHGR for this fuel was 14.5 kW/ft.

**RWE–KAHL.** As of Aug. 31, 1968, the initial core fuel remaining in the KAHL reactor had attained an average exposure of 11,000 MWd/metric ton on 1368 segments with a peak exposure of 21,400 MWd/metric ton. The peak operating LHGR for this fuel was 13.0 kW/ft. Only two leaker assemblies have been reported, but visual examination did not reveal any cladding perforations.

**Humboldt Bay.** There are 6468 Humboldt Type II fuel rods currently operating in the Humboldt Bay reactor. The average exposure of these segments as of Dec. 31, 1969, was 11,700 MWd/metric ton with estimated peak rod and pellet exposures of 16,900 and 20,300 MWd/metric ton, respectively. The peak operating LHGR for this fuel was 12.1 kW/ft. No known

failures had occurred in the Type II fuel, and no leakers had been detected. In addition to Type II, there are currently 1872 Type III Zircaloy-clad $UO_2$-pellet fuel rods (loaded March 1969) operating in the Humboldt Bay reactor. As of Dec. 31, 1969, the Type III fuel had achieved an average exposure of 2500 MWd/metric ton with peak rod and pellet exposures of 4200 and 5060 MWd/metric ton, respectively, and a peak LHGR of 16.8 kW/ft.

**KRB.** Of the 368 fuel assemblies (13,248 Zircaloy-clad fuel rods) initially loaded in 1966, 292 (10,512 fuel rods) are currently operating. As of Dec. 31, 1969, the fuel had attained an average exposure of ~10,100 MWd/metric ton with peak rod and pellet exposures estimated to be ~14,200 and ~17,100 MWd/metric ton, respectively, for a peak operating LHGR of 15.8 kW/ft. During a scheduled outage in July 1967 for a planned equipment modification, almost the entire core was sipped. Approximately 10% of the assemblies gave some indication of leaking, i.e., a signal higher than background. Three of the assemblies with high readings were removed for borescope and ultrasonic inspection in the fuel pool and replaced with fresh fuel. The other assemblies were returned to the reactor for continued irradiation. Failure in one rod of each of these assemblies was located. The reason for these failures was fretting wear of the fuel cladding caused by the presence of foreign material (i.e., wires) found in the assemblies. The wires came from the wire demister section of the steam separators; the steam-separator design has now been changed for Dresden 2 and other modern BWRs to eliminate the use of wires in the demister.

During the planned 1969 summer refueling outage, almost the entire core loading was sipped, as in 1967. Approximately 20% of the fuel assemblies gave some indication of leaking, i.e., a signal higher than background. These assemblies, as well as several sound assemblies, were examined visually and tested ultrasonically at the site to locate possible defects. The detailed examinations revealed a total of 127 fuel rods, out of the 13,140 in the reactor, with in-service failures. These 127 rods plus the 3 defective rods detected in 1967 make a total of 130 in-service failures in the original core loading of 13,248 fuel rods, i.e., <1%. It is significant that the majority of the fuel bundles that contained failed fuel rods have been repaired at the site for return to service by removal and/or replacement of defective rods.

Analysis of the visually observed failures after location by sipping and ultrasonic testing revealed three distinct categories of cladding perforation. Approximately one-half of the failures were similar to those observed in 1967, i.e., fretting wear caused by steam-separator wires. The other one-half of the observed failures were divided between tube flaws introduced during fabrication and attack by internal impurities, both of which resulted from deviations from previous manufacturing practices. These deviations were investigated and corrected so as to preclude, on that scale, initial tubing defects and failures from internal impurities in reactors of the Browns Ferry 1 class. However, a few failures resulting from tube flaws and impurities that occur statistically in any production-quantity fabrication process should not be unexpected. As indicated above, recurrence of the fretting-wear failures is not expected in future systems because of elimination of the use of wires in the demister section of the steam separator.

**Consumers Power Co. (Big Rock Point).** A number of fuel types have operated in the Big Rock Point reactor since initial startup in 1962. As of Dec. 31, 1969, the Type B Zircaloy-clad $UO_2$-pellet fuel design has accumulated the highest exposures of the currently operating designs in Big Rock Point; 1694 fuel rods are currently operating with an average exposure of ~19,500 MWd/metric ton, peak rod and pellet exposures estimated to be ~24,500 and ~29,500 MWd/metric ton, respectively, and a peak operating LHGR of 15.0 kW/ft. Prior to April 1969 there were no known failures and two suspected leakers in the Type B fuel assemblies.

As of Dec. 31, 1969, the Type E reload fuel (loaded in July 1968) had accumulated an average of ~6600 MWd/metric ton on 38 assemblies (2926 fuel rods) with peak rod and pellet exposures estimated to be 11,200 and 13,400 MWd/metric ton, respectively, and a peak design LHGR of 17.7 kW/ft.

As of Dec. 31, 1969, the Type E-G reload fuel (loaded in April 1969) had accumulated an average of 3500 MWd/metric ton on 22 assemblies (1694 fuel rods) with peak rod and pellet exposures estimated to be 4400 and 5300 MWd/metric ton, respectively, and a peak design LHGR of 17.7 kW/ft.

During the refueling outage of April 1969, some of the fuel assemblies in the core were checked by the dry sipping technique to find leakers. The sipping revealed four leakers in the Type B fuel and three in the Type E fuel. Further inspection of two Type B and two Type E leakers revealed five and nine failed rods, respectively. In addition, there were eight failed fuel rods observed in the centermelt development fuel

assembly D-50. The observed failures were of the same character in all fuel types inspected and were limited to no more than 20 in. of active fuel length in any given fuel rod. Observations and analyses made to date indicate that the fuel-rod failures resulted from the heavy buildup of crud scale that caused the cladding surfaces to overheat to abnormally high temperatures. At the peak-power locations, the cladding surfaces corroded at the high rates expected at those high temperatures. The crud resulted from the materials in the feedwater-condensate system. The design selection of materials in the feedwater system and the design of condensate demineralizer equipment for the modern BWRs have been based on the requirement to stringently limit impurities in the reactor water to preclude the occurrence of this type of failure.

**JPDR.** The Japan Power Demonstration Reactor (JPDR) has been operating since December 1963 with a core load of Zircaloy-clad $UO_2$-pellet fuel. The final 72 fuel assemblies of the initial core load (5184 fuel rods) were discharged in 1969 with an average exposure of 8200 MWd/metric ton. Peak operating LHGR for this fuel was 13.0 kW/ft. The operation of this fuel has been very successful with no reported failures.

**Tarapur.** The Tarapur twin reactors achieved initial operation in May 1969. A total of 568 fuel assemblies (27,832 fuel rods) are operating at this time. As of Dec. 31, 1969, the initial core loading of the two reactors was operating with an average exposure of ~1400 MWd/metric ton and peak rod and local exposures of 2250 and 2700 MWd/metric ton, respectively. The peak design LHGR for the fuel is 15.8 kW/ft with design exposures of 15,000 MWd/metric ton (average) and ~33,000 MWd/metric ton (peak).

## Experience in Fuel-Development Programs[1]

In addition to the experience with production fuel in operating BWRs, much experience has been gained over the past 10 to 12 years in development programs in which many prototype fuel rods and capsules* have been irradiated. A great deal of this development experience is pertinent to the support of the 17.5 to 18.5 kW/ft operating limits and the 28 kW/ft (fresh fuel) damage limit. A summary of the more significant development experience follows.

*A "capsule," as used herein, refers to a test fuel rod, or group of rods combined, with all features similar to production fuel rods except for having reduced active fuel length (as low as ~6 in.).

**General Electric Company.** Tables 9 and 10 are summaries of design details and performance conditions for some fuel rods and capsules, respectively, irradiated under General Electric development test programs. These data complement the BWR production-fuel experience by providing additional data at high power and exposure. The ranges of peak performance conditions covered include:

Exposure, 1500 to 75,000 MWd/metric ton.
LHGR, 13.0 to 58.0 kW/ft.
Heat flux, 400,000 to 1,355,000 Btu/(hr)(ft$^2$).

*Vallecitos Boiling Water Reactor (VBWR).*[4,15] Several prototype fuel rods were operated in the VBWR at thermal-performance conditions slightly exceeding the design conditions for current BWR Zircaloy-clad $UO_2$-pellet fuel. Nine fuel rods were operated at ~19.9 kW/ft and at a heat flux of 460,000 Btu/(hr)(ft$^2$) just prior to the termination of irradiation at 9500 MWd/metric ton (average) and 12,000 MWd/metric ton (peak). Subsequent detailed hot-cell examination of these fuel rods showed completely satisfactory performance.

Under the AEC Fuel Cycle Program [USAEC Contract AT9 (04-3)-189, Project Agreement 11], 144 fuel rods were irradiated[15] in the VBWR at slightly lower, 16.6 kW/ft maximum, LHGRs than for the 9 rods mentioned above and in some cases at even higher surface heat fluxes, 509,000 Btu/(hr)(ft$^2$), to an average exposure of 7400 MWd/metric ton and a peak exposure of 13,800 MWd/metric ton. The fuel rods for the SA-1 assembly were taken from this group of fuel rods when the VBWR operation was terminated. The SA-1 is composed of 98 of these rods which operated satisfactorily in the Dresden 1 reactor to a very long exposure, ~40,000 MWd/metric ton (peak). The maximum LHGR in Dresden 1 operation was 13.0 kW/ft. Ten of these fuel rods were removed for detailed hot-cell examination[4] in 1967. The results of these examinations indicate very satisfactory performance for this fuel. It is interesting that fission-gas measurements and rod-profile measurements indicate lower fission-gas release ($\leq$1.0%) and lower fuel expansion than are allowed for in the current BWR design models for this fuel. The SA-1 fuel assembly, with 88 remaining fuel rods, was finally discharged without apparent failure from the Dresden 1 reactor at the end of cycle 6 during October 1969.

*General Electric Test Reactor (GETR).*[6,16] A large number of test fuel rods and capsules have been irradiated in the GETR. Extensive experience with

**Table 9 General Electric Developmental Irradiations of Zircaloy-Clad High-Density (95% of Theoretical) $UO_2$-Pellet Fuel Rods***

| Name | Reactor | Number of rods | Fuel-rod diameter, in. | Cladding wall thickness, in. | Pellet-to-cladding gap, mils | Peak heat flux, Btu/(hr)(ft$^2$) | Peak LHGR, kW/ft | Peak exposure, MWd/metric ton | Status |
|---|---|---|---|---|---|---|---|---|---|
| Dresden prototype | VBWR | 9 | 0.565 | 0.030 | 3.0–16.0 | 460,000 | 19.94 | 12,000 | Completed |
| Fuel cycle (R&D)† | VBWR | 144 | 0.424 | 0.022 | 2.0–8.0 | 509,000 | 16.6 | 13,800 | Completed |
| Dresden prototypes | VBWR | 52 | 0.565 | 0.028 | 5.0–8.0 | 407,000 | 17.64 | 10,000 | Completed |
| High performance $UO_2$§ | GETR | 12 | 0.565 | 0.030 | 4.0–6.0 | 630,000<br>1,126,000 | 27.0<br>49.0 | 1,500 | Completed‡ |
| High performance $UO_2$§ | GETR | 2 | 0.565 | 0.030 | 4.0–11.0 | 1,355,000 | 58.0 | 14,000 | Completed¶ |
| SA-1** | Dresden 1 | 98 | 0.424 | 0.022 | 4.0–8.0 | 400,000 | 13.0 | 40,000 | Completed |
| D-1, -2, -3†† | Consumers | 363 | 0.424 | 0.030 | 7.0 | 434,000 | 14.2 | 30,000 | Completed |
| D-50‡‡ | Consumers | 36 | 0.570 | 0.035 | 12.0 | 507,000 | 22.0 | 15,400 | Under evaluation§§ |
| D-52, -53 | Consumers | 58 | 0.700 | 0.040 | 13.0 | 525,000 | 27.0 | 4,600 | Under evaluation |

*From Ref. 1.
†USAEC Contract AT(04-3)-189 Project Agreement 11.
‡One rod failure at 49 kW/ft.
§USAEC Contract AT(04-3)-189 Project Agreement 17.
¶Hollow pellet.
**USAEC Contract AT(04-3)-189 Project Agreement 41.
††USAEC Contract AT(04-3)-361.
‡‡USAEC Contract AT(04-3)-189 Project Agreement 50.
§§Eight fuel rods failed during second operating cycle owing to abnormal crud and scale deposition.

**Table 10 General Electric Developmental Irradiations of Zircaloy-Clad High-Density (95% of Theoretical) $UO_2$-Pellet Capsules in the GETR***

| Capsule | Number of rods | Fuel-rod diameter, in. | Cladding wall thickness, in. | Pellet-to-cladding gap, mils | Peak heat flux, Btu/(hr)(ft$^2$) | Peak LHGR, kW/ft | Peak exposure,† MWd/metric ton | Status |
|---|---|---|---|---|---|---|---|---|
| A | 3 | 0.425 | 0.024–0.032 | 1.4–10.2 | 750,000 | 24.5 | 77,000 | Complete |
| | 1 | 0.488 | 0.032 | 11.2 | 785,000 | 29.4 | 34,000 | Complete |
| B | 6 | 0.489 | 0.034 | 7.7–11.6 | 503,000 | 18.8 | 41,000 | Continuing |
| C | 5 | 0.557 | 0.036 | 2.0–15.0 | 404,000 | 17.7 | 34,000 | Continuing‡ |
| D | 5 | 0.557 | 0.036 | 2.0–1.2 | 540,000 | 23.0 | 37,000 | Complete |

*From Ref. 1.
†Peak exposure status is as of 11/30/69.
‡Two sound rods removed for detailed remote-materials-laboratory examination with ~30,000 MWd/metric ton exposure.

3-ft-long Zircaloy-2-clad $UO_2$-solid-pellet fuel rods, operating with gross central $UO_2$ melting and including the effects of BWR environment, power cycling, and irradiation up to 14,000 MWd/metric ton, has been obtained under USAEC Contract AT(04-3)-189, Project Agreement 17, High Performance $UO_2$ Programs.[6,16] Twelve fuel rods were irradiated at an average exposure of ~1500 MWd/metric ton and a maximum LHGR of 27 to 49 kW/ft (630,000 to 1,126,000 Btu/(hr)(ft$^2$)). A single rod failure was believed to have occurred at 49 kW/ft, as evidenced by an increase in noble-gas activity detected just prior to shutdown of the reactor. However, subsequent detailed examination of the fuel rods did not uncover the suspected cladding perforation. Two additional Zircaloy-2-clad hollow $UO_2$-pellet fuel rods were operated to a peak exposure of ~14,000 MWd/metric ton and a peak LHGR of 58 kW/ft [1,355,000 Btu/(hr)(ft$^2$)]. Detailed postirradiation examination indicated completely satisfactory performance for these fuel rods.

In an investigation of the effects of exposure at high power generation, several capsules containing 6-in.-long Zircaloy-clad $UO_2$-pellet fuel rods with characteristics similar to current BWR fuel designs are being irradiated in the GETR to exposures as high as 77,000 MWd/metric ton and peak LHGRs up to 29.4 kW/ft (Table 10). The detailed design features of the Zircaloy-clad $UO_2$-pellet fuel rods within these capsules are given in Table 10. Capsule A contains four Zircaloy-clad $UO_2$-pellet fuel rods with an average exposure of ~54,000 MWd/metric ton and a peak exposure of ~77,000 MWd/metric ton. Continuous operation for extended periods at peak LHGRs as high as 24.5 to 29.4 kW/ft has been achieved at high exposures. The peak rod in capsule A was discharged after 77,000 MWd/metric ton. The remaining rods attained a peak exposure in excess of 60,000 MWd/metric ton. Three Zircaloy-clad $UO_2$-pellet fuel rods originally in capsule A were discovered to have pinhole leaks at approximately 26,000 to 27,000 MWd/metric ton. These fuel rods have received detailed remote-materials-laboratory examination, but the cause of the pinholes could not be determined. Two typical performance histories for fuel rods in GETR capsules are given in Fig. 10. The histories shown are for fuel rods having the highest power and highest exposure, respectively, in capsule A. It is quite significant that very high power-generation rates (18.0 to 29.4 kW/ft LHGR)

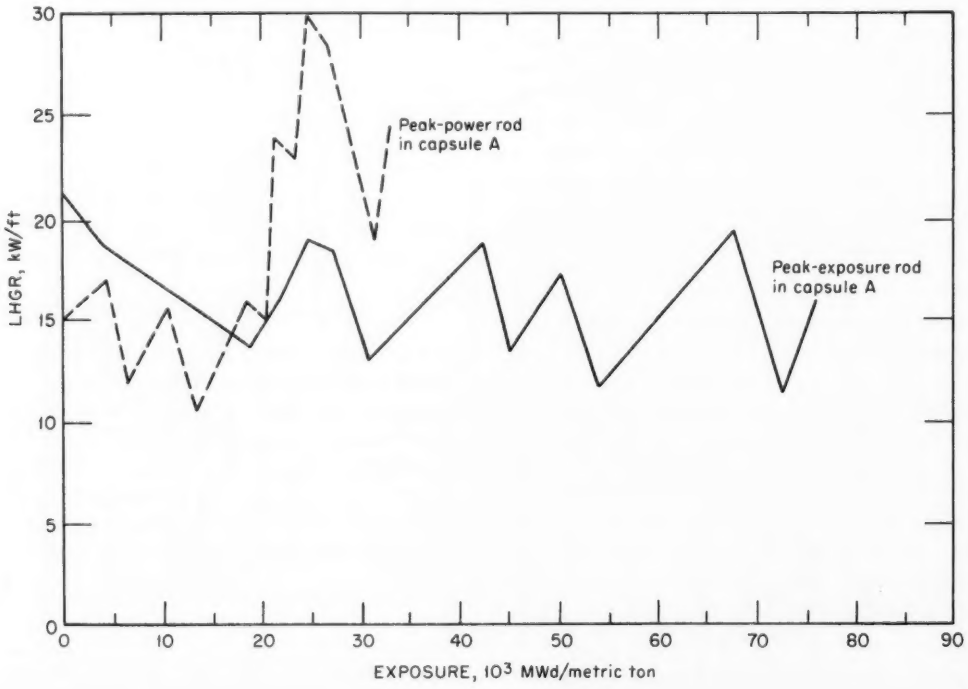


**Fig. 10 Typical LHGR history of GETR capsules (from Ref. 1).**

have been achieved at relatively high exposures (20,000 to 77,000 MWd/metric ton).

Capsule B contains six (and capsules C and D five each) Zircaloy-clad $UO_2$-pellet fuel rods with peak exposures from 34,000 to 41,000 MWd/metric ton and peak LHGRs from 17.7 to 23.0 kW/ft. The results of nondestructive examinations performed to date indicate very satisfactory performance of the GETR capsules. Fuel and cladding dimensional changes are, in general, less than would be predicted by current BWR design models and thus demonstrate that these models exhibit adequate conservatism. No failures have been observed at the conditions anticipated for normal operation in current BWR designs, i.e., $\leq$18.5 kW/ft. However, the capsules were not designed for continuous operation at higher kilowatts per foot, and, when they were operated continuously for extended periods above the design conditions, in-service failures were observed. It is significant that, for short-term operation to a linear power in excess of 28 kW/ft with a burnup of more than 25,000 MWd/metric ton, no fuel damage has been observed. Only with extended continuous operation at conditions above 18.5 kW/ft or with repeated cyclic melting of the $UO_2$ fuel have capsule failures been observed.

Fission-gas analysis of sound fuel rods operated beyond 50,000 MWd/metric ton with a peak LHGR in excess of 20 kW/ft (volume average fuel temperatures as high as ~2400°F) yielded release values of <15%. For these conditions the BWR design model described previously would predict fission-gas releases >35%.

*Dresden 1* (Ref. 4). A limited number of what would be considered high-performance (high exposure or power, or both) developmental fuel assemblies have been irradiated in the Dresden 1 reactor core. The SA-1 fuel assembly with a peak exposure of approximately 40,000 MWd/metric ton was discharged after more than 5 years of successful operation[4] in the Dresden reactor [USAEC Contract AT(04-3)-189, Project Agreement 41]. Initial operation of these fuel rods at a peak LHGR of 16.0 kW/ft was achieved in the VBWR. The maximum LHGR achieved in Dresden operation was ~13.0 kW/ft early in life. Just prior to final discharge from Dresden 1, the SA-1 fuel assembly, with 88 of the original rods, was operating at a peak LHGR of ~6.5 kW/ft.

*Consumers Power Co. (Big Rock Point).*[17,18] A number of high-performance (high exposure or power, or both) developmental fuel assemblies have been operated in the Big Rock Point reactor core. The highest exposure developmental assembly irradiated in Big Rock Point was bundle D-3, which had a peak discharge exposure of ~30,000 MWd/metric ton. The maximum LHGR at any time during life was 14.2 kW/ft for D-3. Bundle D-3 is one of three Big Rock Point developmental fuel assemblies[17] with Zircaloy-clad $UO_2$ pellets irradiated under USAEC Contract AT(04-3)-361. Bundles D-1 and D-2 were previously discharged in 1968 with ~18,000 MWd/metric ton average exposure. Bundle D-2 was detected as a leaker at the time of discharge during the February–March 1968 Big Rock Point outage. However, no positive visual evidence of failure was uncovered during detailed visual examinations at the site. With the exception of this leaker indication, all evidence indicates very satisfactory performance of these fuel assemblies.

The highest power developmental assemblies[18] with Zircaloy-clad $UO_2$-pellet fuel which have operated in Big Rock Point are the centermelt bundles D-50, D-52, and D-53, irradiated under USAEC Contract AT(04-3)-189, Project Agreement 50. Maximum LHGRs of 22.0 to 27.0 kW/ft were achieved by these bundles. The peak burnups were ~15,400 MWd/metric ton for D-50 and ~4,600 MWd/metric ton for D-52 and D-53. Fuel assembly D-50 operated for two cycles prior to being discharged at the April 1969 refueling outage as a consequence of failure. The number and character of the failures within D-50 are similar to the failures (described earlier under "Experience with Operating Production Fuel") observed in the type B and E reload fuels discharged at this same time. Note that these observed failures resulted as a consequence of abnormally high crud deposits and are not indicative of the performance capability of BWR fuel operated in the Dresden 2 reactor and other modern BWR plants. To the contrary, when the performances of bundles D-50, D-52, and D-53 are considered as a whole, they serve to demonstrate that BWR Zircaloy-clad $UO_2$ fuel can be designed to operate for extended periods under very adverse conditions. Fuel assemblies D-52 and D-53 operated successfully without failure for one cycle and then were discharged for inspection.

*OECD Halden Reactor.*[19-22] The Halden irradiations have successfully demonstrated fuel-rod operation up to a peak LHGR of 24.8 kW/ft at a peak exposure of 16,500 MWd/metric ton (IFA-29). A few rod failures have been experienced, but these may be attributed to inadequate pellet-to-cladding gaps resulting in excessive long-term cladding strain and/or apparent internal corrosion related to possible inadequate control of moisture during fabrication. From the standpoint of fuel thermal performance attained with a

Zircaloy-clad $UO_2$-pellet fuel design, this experience is significant. The only failures observed are apparently due to mechanisms that are now understood and prevented by adequate design allowance for relative expansion of fuel and cladding and by proper control of fabrication impurities in the Zircaloy-clad $UO_2$-pellet fuel employed in the modern BWR. A summary of the Halden experience follows.

*United Kingdom.* The British irradiated 10 five-rod BWR fuel bundles in Halden; most with Zircaloy-2 cladding but some with Zircaloy-4 and Zr–Nb (Table 11). All the fuel was sintered-pellet $UO_2$ with an overall fuel-column length of 60 in. The cladding had a $^{9}/_{16}$-in. inside diameter and 14- and 18-mil walls (500-psi coolant pressure). The most successful test was IFA-29, which operated to 12,000 MWd/metric ton average bundle burnup at a peak heat rating of 24.8 kW/ft (LHGR). Bundle IFA-22 operated to 6600 MWd/metric ton average burnup (8400 peak) at 18.9 kW/ft but experienced a rod failure due apparently to internal hydriding.[23] Three other bundles had similar rod failures. All failures were attributed to internal contamination of the tubes with fluoride coupled with excessive moisture in the $UO_2$, a mechanism that is now understood and currently prevented by control of impurities in the fuel rods of modern BWRs.

*Halden Project.* Over 50 bundles have been irradiated to date, nearly all with Zircaloy-2 cladding and sintered-pellet $UO_2$ fuel. Each bundle usually contains five rods in a cluster with the rods having an outside diameter of ~0.600 in. and a fuel-column length of 60 in. The Halden reactor uses heavy water as the coolant with an operating pressure of 400 to 500 psi. At least 20 bundles have exceeded 5000 MWd/metric ton, and two have passed 10,000 MWd/metric ton. Bundle IFA-29 of the United Kingdom's Atomic Energy Research Establishment is the star performer, but bundle IFA-4 of the Halden project also has reached 12,000 MWd/metric ton with a peak rating of ~8 kW/ft (LHGR). Bundle IFA-11 (Swedish) has reached 10,000 MWd/metric ton with a peak rating of 15 kW/ft (LHGR). Table 11 summarizes the high-power Zircaloy-clad-pellet fuel tests.

Only two bundle failures have been documented: IFA-137 (Italian) and IFA-21 (Swedish). Bundle IFA-137 failed after a very short period of irradiation (<500 MWd/metric ton) with multiple cladding cracks and hydride blisters (sunbursts), apparently due to internal cladding corrosion related to possible inadequate control of moisture during fabrication. Bundle IFA-21 failed with a peak exposure of ~12,200 MWd/metric ton. The failure was attributed to inadequate pellet-to-cladding gap resulting in excessive long-term cladding strain from relative expansion of fuel and cladding. Both mechanisms are understood, and appropriate design specifications are employed to preclude them in modern BWR fuel rods.

**Battelle Memorial Institute.** *Plutonium Recycle Test Reactor (PRTR).*[24] Approximately 1200 compacted-powder fuel rods, 36 hot-pressed-pellet fuel rods, and 2 cold-pressed-and-sintered-pellet fuel rods have been irradiated in the PRTR at peak powers up to 21 kW/ft (LHGR) and peak exposures of 10,000 MWd/metric ton. All these fuel rods contained $UO_2$–$PuO_2$ fuel. The pellet fuel assemblies have operated successfully without the occurrence of failure. One of the bundles containing pellet fuel rods did give an indication of a pinhole leak; however, sipping tests failed to locate any defect. Tests of the PRTR pellet rods are summarized in Table 12.

**Westinghouse Electric Corp.** *Westinghouse Commercial Atomic Power Tests (Saxton).*[2,25,26] The regular Saxton core II operation was terminated on Oct. 18, 1968, and a special test was begun to evaluate the ability of mixed oxide ($UO_2$–$PuO_2$) to withstand short-duration overpower operation after moderately high burnups. The Zircaloy-4-clad mixed-oxide fuel in Saxton II had reached a core average burnup of 17,400 MWd/metric ton of metal, a peak rod burnup of 25,250 MWd/metric ton of metal, and a peak pellet burnup of 33,800 MWd/metric ton of metal (Table 13).

Two sintered-pellet Zircaloy-4-clad rods with average burnups of 13,400 and 13,800 MWd/metric ton of metal were selected for the overpower test. Each rod had a 0.392-in. outside diameter and a 23-mil-thick cladding and had operated previously at 10 to 11 kW/ft. The fuel was preconditioned by operation for 7 days at 19 kW/ft and then removed for examination. The rods showed some increase in ovality (0.2 to 0.6 mil) from that prior to conditioning. The overall conclusion was that there was some additional creep-down (creep collapse) of the cladding from the conditioning operation. After reinsertion the rods were restabilized at the level of ~19 kW/ft, were then subjected to a rapid power increase (2 min) to 21.5 kW/ft, and were held at this high power for 5 days, after which the reactor was shut down. Subsequently the same rods were subjected without difficulty to three similar overpower steps. Postirradiation evaluation showed no apparent damage to the rods from the transients.

Table 11 Tests of Zircaloy-Clad $UO_2$-Pellet Fuel in the Halden Heavy-Water BWR with ~400-psi Coolant Pressure*

| Source and original reference | Assembly No. | Number of rods | Pellet-to-cladding gap, mils | Wall thickness, in. | Outside diameter, in. | Length, in. | Peak heat flux, Btu/(hr)(ft$^2$) | Peak LHGR, kW/ft | Peak exposure, MWd/metric ton | Comments |
|---|---|---|---|---|---|---|---|---|---|---|
| UKAEA[21] | IFA-6 | 5 | 3.5–6.5 | 0.019 ± 0.002 | 0.613 | 65.7 | 350,000 | 17.7 | 4300 | Failed |
| UKAEA[21] | IFA-7 | 5 | 3.5–6.5 | 0.019 ± 0.002 | 0.613 | 65.7 | 350,000 | 17.7 | 4500 | |
| UKAEA[21] | IFA-8 | 5† | 3.5–6.5 | 0.019 ± 0.002 | 0.613 | 65.7 | 350,000 | 17.7 | 4840 | Failed |
| UKAEA[21] | IFA-19 | 5† | 3.5–6.5 | 0.019 ± 0.002 | 0.613 | 65.7 | 350,000 | 17.7 | | |
| | | | 8.5–11.5 | 0.019 ± 0.002 | 0.613 | 65.7 | 350,000 | 17.7 | 3500 | Failed |
| UKAEA[21] | IFA-20 | 5† | 3.5–6.5 | 0.019 ± 0.002 | 0.613 | 65.7 | 350,000 | 17.7 | 5600 | |
| UKAEA[21] | IFA-22 | 5 | 3.5–6.5 | 0.019 | 0.613 | 65.7 | 350,000 | 17.7 | | |
| | | | | 0.014 | 0.613 | 65.7 | 370,000 | 18.9 | 8400 | Failed |
| UKAEA[21] | IFA-23 | 5 | 3.5 | 0.019 | 0.613 | 65.7 | 370,000 | 18.9 | 6300 | |
| | | | | 0.014 | 0.613 | 65.7 | 370,000 | 18.9 | | |
| UKAEA[21] | IFA-28 | 5† | 3.5 | 0.020 | 0.613 | 65.7 | 370,000 | 18.9 | 5300 | |
| | | | | 0.015 | 0.613 | 65.7 | 370,000 | 18.9 | | |
| UKAEA[21] | IFA-29 | 5 | 3.5 | 0.019 | 0.613 | 65.7 | 495,000 | 24.8 | 16,500 | |
| UKAEA[21] | IFA-41 | 5 | 3.5 | 0.017 | 0.61 | 65.7 | 370,000 | 18.9 | 800 | |
| | | | | 0.020 | | 65.7 | | | | |
| Norway[21,22] | IFA-4 | 14 | 4.3–10.0 | 0.018 | 0.54 | 31.4 | 160,000 | 8.0 | 15,400 | |
| Sweden[21] | IFA-10 | 6 | 6 | 0.025 | 0.547 | 67.5 | 314,000 | 14.1 | 4200 | |
| Sweden[21] | IFA-11 | 5 | 2.0–6.0 | 0.021–0.025 | 0.55 | 67.5 | 365,000 | 16.4 | 9900 | |
| Sweden[19] | IFA-21 | 5 | 2.0–7.0 | 0.019–0.025 | 0.55 | 67.5 | 300,000 | 15.2 | 12,200 | Failed |
| Germany[21] | IFA-26 | 7 | 5.5–12.0 | 0.024 | 0.596 | 43.5 | 435,000 | 21.5 | 6600 | |
| Italy[20] | IFA-137 | 21 | | 0.020&0.039 | 0.786 | | 281,000 | 16.9 | 320 | Failed |

*From Ref. 1.

†Assemblies contain some (1 to 4) Zr–Nb clad rods.

**Table 12 Data on Tests by Battelle–Northwest on Zircaloy-Clad Pellet Fuel Rods in the PRTR with 1050-psi Coolant Pressure***

| | | | |
|---|---|---|---|
| Source | Battelle–Northwest | Peak LHGR, kW/ft | 21 |
| Number of rods | 38 | Peak exposure, MWd/metric ton | 10,000 |
| Active fuel-column length, in. | 58.5 | Comments | $UO_2$–$PuO_2$ pellets; indication of a possible pinhole failure |
| Wall thickness, in. | 0.030 | | |
| Cladding to fuel clearance, mils | 12 | | |
| Outside diameter, in. | 0.565 | | |
| Peak heat flux, Btu/(hr)($ft^2$) | 485,000 | | |

*Taken from Ref. 1. Material originally from Ref. 24.

**Table 13 Westinghouse Data on Zircaloy-2- and Zircaloy-4-Clad Fuel from Loop Tests in the WTR and from Tests in the Saxton Reactor with 2000-psi Coolant Pressure***

| Reactor (original reference) | Number of rods | Fuel-column length, in. | Cladding thickness, in. | Outside diameter, in. | Pellet-to-cladding gap, mils | Peak $Q/A$, Btu/(hr)($ft^2$) | Peak LHGR, kW/ft | Peak exposure, MWd/metric ton | Comments |
|---|---|---|---|---|---|---|---|---|---|
| WTR[25] | 3 | 38 | 0.032 | 0.500 | 2, 6, and 12 | 495,000 | 19.0 | 3,450 | |
| WTR[25] | 4 | 6 | 0.032 | 0.500 | 2, 6, and 12 | 576,000 | 22.2 | 6,300 | |
| Saxton[26] | 58 | 36 | 0.023 | 0.391 | 6.5 | 550,000 | 16 | 17,050 | Zircaloy-2- and Zircaloy-4-clad fuel; three intentionally defected rods performed well |
| Saxton[2,26] | 600 | 36 | 0.023 | 0.391 | 7.0 | 550,000 | 16 | 33,800 | Mixed oxide ($UO_2$–$PuO_2$) fuel. Most rods have powder fuel |
| Saxton[26] | 2 | 36 | 0.023 | 0.391 | 7.0 | 740,000 | 21.5 | ~13,800 | Mixed oxide ($UO_2$–$PuO_2$) fuel |

*From Ref. 1.

*Westinghouse Test Reactor (WTR).*[25] Three full-length rods and four short pins were irradiated in the WTR to 6300 MWd/metric ton peak exposure at LHGRs up to 22 kW/ft (Table 13). The performance of these fuel rods was entirely satisfactory (Table 13).

*Carolinas–Virginia Tube Reactor (CVTR).*[27-29] The CVTR irradiation experience with Zircaloy-clad $UO_2$-pellet fuel is summarized in Table 14. A total of 1368 standard fuel rods attained a peak exposure of 19,800 MWd/metric ton and a peak LHGR of 13.6 kW/ft. Two special fuel assemblies achieved LHGRs in excess of 20 kW/ft with peak exposures in excess of 10,000 MWd/metric ton. Only five failures were detected in the total of 1390 fuel rods (<0.5% failures). Each of three special assemblies had one fuel-rod failure. Two of these rods had freestanding Zircaloy-4 cladding and one had thin-wall Zircaloy-4 cladding. In addition, a single rod failed in each of two standard assemblies. The failures generally were characterized by multiple cladding defects and localized internal hydriding, but no correlation could be found in terms of design or irradiation performance to account for them. Westinghouse analysis of these failures is as follows: "The evidence suggests that the failures were caused by internal sources of hydrogen which led to local hydriding of the Zircaloy clad. The combination of clad stresses and local hydrogen embrittlement then produced the observed failures. The internal hydriding is most likely the result of contamination of fuel rod components during fabrication."

**Atomic Energy of Canada Limited (AECL) (Chalk River).**[30,31] Essentially all the AECL work has been concentrated on Zircaloy-2- or Zircaloy-4-clad $UO_2$-sintered-pellet fuel for applications in pressurized heavy-water reactors (PHWRs) or boiling light-water reactors (BLWRs). The fuel rods are generally ~20 in.

**Table 14 Westinghouse Data on Zircaloy-4-Clad Fuel from Tests in the CVTR with 2000-psi Heavy Water***

| Assembly No. | Rod No. | Cladding thickness, in. | Outside diameter, in. | Pellet-to-cladding gap, mils | Peak $Q/A$, Btu/(hr)(ft$^2$) | Peak LHGR, kW/ft | Peak exposure, MWd/metric ton | Comments |
|---|---|---|---|---|---|---|---|---|
| G-1 | 33.831 | 0.0222 | 0.4839 | 6.6 | ~475,000 | ~17.6 | ~8,800 | |
| | W13.831 | 0.0222 | 0.4839 | 6.6 | ~475,000 | ~17.6 | ~8,800 | |
| | 44.731 | 0.0222 | 0.4839 | 7.8 | 570,000 | 21.2 | 10,600 | |
| | 44.732 | 0.0222 | 0.4839 | 7.8 | 570,000 | 21.2 | 10,600 | Failed |
| | 29.732 | 0.0222 | 0.4839 | 7.8 | 570,000 | 21.2 | 10,600 | |
| | 33.833 | 0.0222 | 0.4839 | 6.6 | ~475,000 | ~17.6 | ~8,800 | |
| G-2 | 13.836 | 0.0222 | 0.4839 | 6.6 | 430,000 | 16.0 | 12,500 | |
| | 13.834 | 0.0222 | 0.4839 | 6.6 | 430,000 | 16.0 | 12,500 | |
| | 13.833 | 0.0222 | 0.4839 | 6.6 | 430,000 | 16.0 | 12,500 | |
| | 13.831 | 0.0222 | 0.4839 | 6.6 | 430,000 | 16.0 | 12,500 | |
| | 13.835 | 0.0222 | 0.4839 | 6.6 | 430,000 | 16.0 | 12,500 | |
| | 13.832 | 0.0222 | 0.4839 | 6.6 | 430,000 | 16.0 | 12,500 | |
| G-3 | 53.831 | 0.0222 | 0.4839 | 6.6 | 465,000 | 17.2 | 6,800 | Failed |
| | 13.837 | 0.0222 | 0.4839 | 6.6 | 465,000 | 17.2 | 6,800 | |
| | 64.761 | 0.0222 | 0.4839 | 7.8 | 560,000 | 20.6 | 8,100 | |
| | 64.762 | 0.0222 | 0.4839 | 7.8 | 560,000 | 20.6 | 8,100 | |
| | 24.734 | 0.0222 | 0.4839 | 7.8 | 560,000 | 20.6 | 8,100 | |
| | 53.862 | 0.0222 | 0.4839 | 6.6 | 465,000 | 17.2 | 6,800 | |
| G-4 | 83.832 | 0.016 | 0.4855 | 6.0 | 475,000 | 17.6 | 12,800 | |
| | 13.838 | 0.0222 | 0.4839 | 6.6 | 475,000 | 17.6 | 12,800 | |
| | 13.839 | 0.0222 | 0.4839 | 6.6 | 475,000 | 17.6 | 12,800 | |
| | 83.831 | 0.016 | 0.4855 | 6.0 | 475,000 | 17.6 | 12,800 | Failed |
| Standard assemblies | 1366 rods | 0.02575 | 0.4875 | 6.0 | ~ 370,000 | ~13.6 | ~19,800 | 1366 rods, no failures |
| 502-2-31 | 1549 | 0.02575 | 0.4875 | 6.0 | 330,000 | 12.3 | 18,900 | Failed |
| 502-1-18 | 1475 | 0.02575 | 0.4875 | 6.0 | 374,000 | 13.6 | 19,100 | Failed |

*From Refs. 27 to 29.

long; they are 0.6 to 1 in. in diameter and have nonfreestanding cladding thicknesses between 15 and 22 mils. The rods are arranged in clusters of 7, 19, 28, or 18 rods to fit the pressure-tube inside diameter of the particular test facility and are spaced by either spiral wrap or, more recently, by brazed split spacers. Table 15 summarizes the AECL experience with bundle irradiations.

Substantial irradiation experience (335 bundles, ~5282 rods) has been developed under pressurized-water conditions in the CANDU-class PHWR. This fuel has operated up to a maximum LHGR of ~13 kW/ft and to a maximum exposure of 20,000 MWd/metric ton in the first charge to the NPD reactor. Only five fuel failures have been detected in NPD, and all these have been attributed to maloperation of fuel-loading equipment.

The remainder of the Chalk River experience is from full-scale bundle tests in loops in the NRU at Chalk River at both PWR and BWR conditions. The Canadians report 31 bundle tests (760 rods) at heat ratings up to 24 kW/ft (LHGR) and exposures of 8600 MWd/metric ton at PWR conditions. Under boiling-water conditions the Canadians have tested 54 bundles (1000 rods) at heat ratings up to 29 kW/ft (LHGR) and exposures of 6000 MWd/metric ton. Some of these irradiations are still continuing. The highest heat ratings were reached during short-term tests to investigate the critical heat flux in-pile. Most of the test work is concentrated close to the nominal Gentilly plant rating of 18 kW/ft (LHGR). A number of defects have occurred in the NRU tests. The defects resulted from three causes: (1) improper handling, (2) internal contamination, and (3) gross overpower that caused greater than 1% plastic strain of the cladding.[31]

### Bettis LSBR/LWBR Development Tests[10-14]

The LSBR fuel tests conducted by Bettis Atomic Power Laboratory consisted of approximately 140 rods containing $UO_2$ fuel, $UO_2$–$ZrO_2$ fuel, and $UO_2$–$ZrO_2$–CaO fuel irradiated in loops in the MTR, ETR, and NRX. During these tests they had approximately 14 failures. Table 16 summarizes the results of the long-term irradiations of 18 $UO_2$ fuel rods, of which seven had cladding failures and two others, which were bowed or bulged, were classified as failures.

The failures were generally attributed to excessive cladding strain caused by the combined effects of differential fuel/cladding thermal expansion and fission-product-induced fuel swelling. The LSBR test rods were intentionally fabricated with a very tight (1-mil) gap to accentuate the in-service cladding strains and facilitate deduction of fuel expansion. The lifetime

**Table 15 Data from Canadian (AECL) Irradiation Tests on Zircaloy-Clad Fuel Bundles***

| Experiment | Number of rods | Pellet-to-cladding gap, mils | Wall thickness, in. | Outside diameter, in. | Peak $Q/A$, Btu/(hr)(ft$^2$) | Peak LHGR, kW/ft | Peak exposure, MWd/metric ton |
|---|---|---|---|---|---|---|---|
| Pressurized heavy water† | 38 | ~2–7 | 0.015 | 0.60 | 240,000 | 11 | 19,800 |
| (NPD) at 1150 psi | 19 | ~2–7 | 0.015 | 0.60 | 260,000 | 12 | 12,100 |
| | 4940 | ~2–7 | 0.015 | 0.60 | 175,000 | 8 | 12,100 |
| | 19 | ~2–7 | 0.015 | 0.60 | 295,000 | 13.6 | 11,000 |
| | 19 | ~2–7 | 0.015 | 0.60 | 160,000 | 7.4 | 5,500 |
| | 247 | ~2–7 | 0.015 | 0.60 | 305,000 | 14 | 11,000 |
| Pressurized light water | 228 | ~2–7 | 0.015 | 0.60 | 500,000 | 23 | 8,600 |
| (NRU reactor)‡ at | 532 | ~2–7 | 0.015 | 0.60 | 520,000 | 24 | 5,500 |
| 1450 psi | | | | | | | |
| Boiling light water | 342 | ~2–7 | 0.015 | 0.60 | 520,000 | 24 | 5,800 |
| (BLW-NRU reactor)‡ | 114 | ~2–7 | 0.015 | 0.60 | 390,000 | 18 | 5,000 |
| at 820 psi | 110 | ~2–7 | 0.023 | 0.78 | 520,000 | 31 | 1,000 |
| | 270 | ~2–7 | 0.023 | 0.78 | 485,000 | 29 | 6,000 |
| | 90 | ~2–7 | 0.023 | 0.78 | 420,000 | 25 | 1,200 |

*From Refs. 30 and 31.

†Five defects in NPD tests due to maloperation of fuel-loading equipment.

‡"Number of defects" have occurred in the NRU tests because of "internal contamination of or gross overpower" resulting in greater than 1% cladding strain.[31]

Table 16 Data from Westinghouse–Bettis Tests on Zircaloy-4-Clad 6-in.-Long $UO_2$ Fuel Rods in LSBR/LWBR 2000-psi Pressurized-Water Loops*

| Rod number | Reference | Wall thickness, in. | Outside diameter, in. | Cladding to fuel clearance, mils | Peak $Q/A$, Btu/(hr)(ft$^2$) | Peak LHGR, kW/ft | Exposure, MWd/ metric ton | Pellet shape | % T.D.† stack | Comments |
|---|---|---|---|---|---|---|---|---|---|---|
| 9-16 | 11 | 0.033 | 0.562 | 0.0004 | ~400,000 | 17.6 | 19,300 | Solid | 82 | Failed |
| 79-18 | 10,11 | 0.017 | 0.284 | 0.0004 | 486,000 | 11.0 | 52,300 | Solid | 84 | |
| 79-22 | 7,11 | 0.017 | 0.284 | 0.0004 | 524,000 | 11.4 | 28,000 | Solid | 96.5 | Failed |
| 79-163 | 7,11 | 0.017 | 0.284 | <0.001 | 484,000 | 10.5 | 24,000 | Solid | 97.0 | Bowed |
| 79-121 | 11 | 0.0183 | 0.252 | 0.0036 | ~550,000 | 10.6 | ~22,000 | Solid | ~82 | |
| 79-205 | 11 | 0.034 | 0.5623 | 0.0004 | ~390,000 | 16.8 | 19,200 | Solid | 82 | |
| 79-206 | 11 | 0.016 | 0.285 | 0.0004 | ~430,000 | 9.4 | 26,900 | Solid | 84 | |
| B4-2 | 11 | | 0.410 | 0.001 | | | ~15,400 | Solid | ~95 | Failed |
| B4-2A | 11 | | 0.410 | 0.001 | | | ~52,800 | Solid | ~95 | Failed |
| B4-4A | 11 | | 0.410 | 0.001 | | | ~56,100 | Solid | ~95 | Failed |
| 79-19 | 9 | 0.015 | 0.2816 | 0.0004 | 454,000 | 9 | 44,000 | Annular | 83 | Failed |
| 79-83 | 11 | 0.0183 | 0.2525 | 0.0004 | | | ~20,600 | Annular | ~83 | |
| 79-207 | 11 | 0.015 | 0.2822 | 0.0004 | ~380,000 | 8.2 | 33,100 | Annular | ~83 | |
| 79-209 | 11 | 0.033 | 0.5611 | 0.0004 | ~612,000 | 26.3 | 6,500 | Annular | ~83 | |
| 79-21 | 8 | 0.016 | 0.281 | 0.0004 | ~550,000 | 12.2 | 47,900 | Dished | 85 | Failed |
| 79-25 | 8 | 0.026 | 0.302 | 0.0004 | ~510,000 | 10 | 60,700 | Dished | 85 | Bulged |
| 79-105 | 11 | 0.019 | 0.252 | 0.0038 | | | ~20,900 | Dished | ~85 | |
| 79-208 | 11 | 0.029 | 0.3062 | 0.0004 | ~380,000 | 8.9 | 31,100 | Dished | 85 | |

*From Ref. 1.
†T.D. = theoretical density.

of these fuel rods could possibly have been extended by having a larger cladding–fuel gap to accommodate the fuel swelling. Internal voids in the fuel in the form of sintering porosity or geometric voids, such as dishing as used in Dresden 2 and other modern BWR fuels, generally increased the ability to achieve higher exposures.[11]

## CONCLUSIONS

The basic features of the current BWR fuel design have evolved as a consequence of the satisfactory performance of the Zircaloy-2-clad fuel in general, coupled with data from surveillance of lead production-fuel assemblies, development programs, and data derived from examination and analysis of the fuel failures that have been observed. Commercial operating experience to date with production quantities of fuel has provided statistical demonstration of the design- and manufacturing-process capabilities. The observed mechanisms of cladding failure have led to corrective action in the current design or manufacturing process to reduce the probability of their recurrence. Developmental irradiations have provided demonstrations of local performance conditions, including the peak combinations of local rod power and exposure, which go beyond the requirements for fuel in the modern BWR. In addition, detailed measurements of performance considerations, i.e., fission-product release, cladding strain, etc., have provided a basis for design representation of life-limiting mechanisms. The experience and data obtained support the expectation of not only safe, but reliable, operation of the current BWR Zircaloy-clad $UO_2$-pellet fuel design for the Dresden 2 and Browns Ferry 1 classes of reactors. The relation between operating experience and the proposed performance conditions is illustrated in Fig. 11. The peak and average performances for the maximum-duty fuel pellet in a Browns Ferry 1 fuel rod, including the anticipated peak transient performance, are also shown

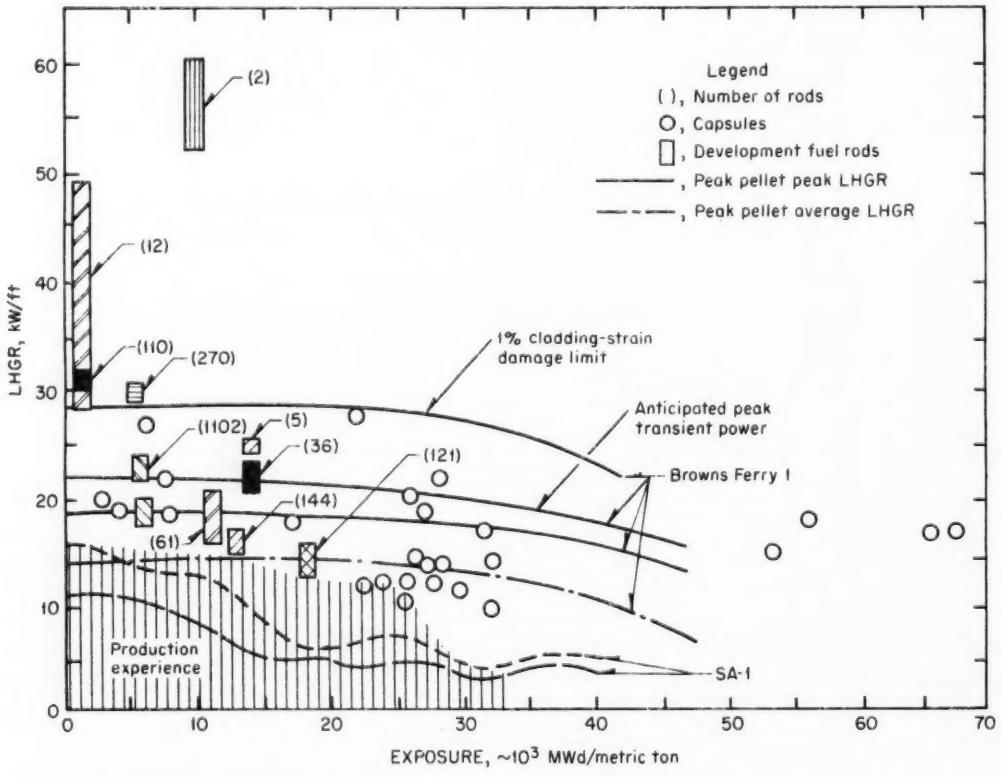


**Fig. 11 Comparison of proposed operation to production and development experience. The different types of crosshatching and filled-in (black) symbols represent data from various development programs.**

as a function of exposure. For comparison, the peak pellet performance at the 1% cladding-strain condition is also shown as a function of exposure.

A value of 1% plastic strain of Zircaloy cladding has been demonstrated as a conservative limit below which fuel damage due to a single transient overstraining is not expected to occur. The LHGR required to cause this amount of cladding strain is calculated to be ~28 kW/ft in the current design. The conservatism of this calculation has been confirmed by a limited number of high-performance test fuel rods.

The continuing irradiation of developmental fuel is to provide a basis for possible future extension in fuel performance. Surveillance of lead performance full-length fuel rods with respect to exposure and LHGRs will continue[32] so that any previously unrecognized performance-limiting phenomena can be identified.

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